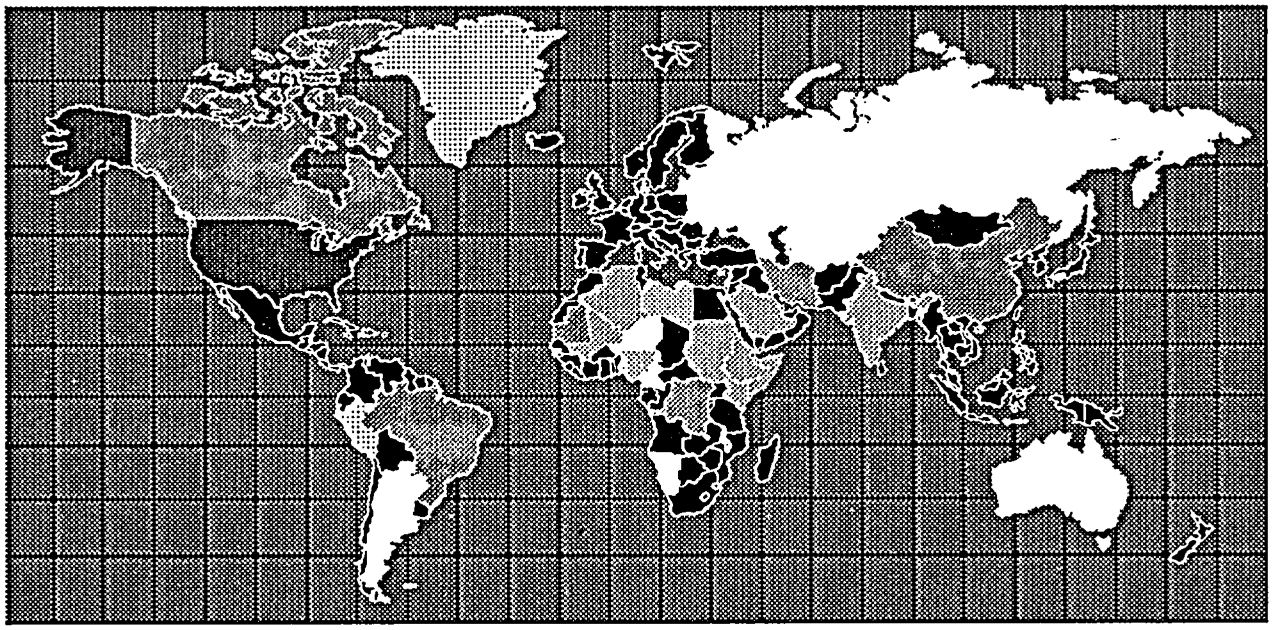
NUREG/CP-0100

Proceedings of the
International Nuclear Power Plant Aging Symposium



Held at
Hyatt Regency
Bethesda, Maryland
August 30–31, 1988 and September 1, 1988

Sponsored by
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

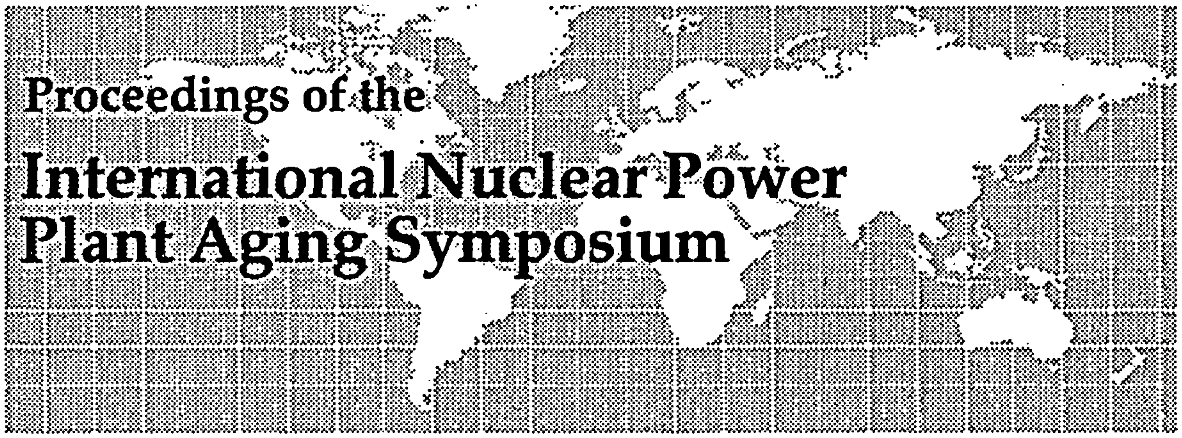
Available from

Superintendent of Documents
U.S. Government Printing Office
P.O. Box 37082
Washington D.C. 20013-7082

and

National Technical Information Service
Springfield , VA 22161

NUREG/CP-0100



Proceedings of the International Nuclear Power Plant Aging Symposium

Held at
Hyatt Regency
Bethesda, Maryland
August 30–31, 1988 and September 1, 1988

Date Published: March 1989

Compiled and Edited by A. Beranek

Sponsored by
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

In Cooperation with

American Nuclear Society

American Society of Civil Engineers

American Society of Mechanical Engineers

Institute of Electrical and Electronics Engineers

For sale by the Superintendent of Documents, U.S. Government Printing Office
Washington, D.C. 20402

Abstract

This report presents the proceedings of the International Nuclear Power Plant Aging Symposium that was held at the Hyatt Regency Hotel in Bethesda, Maryland, on August 30-31 and September 1, 1988. The Symposium was presented in cooperation with the American Nuclear Society, the American Society of Civil Engineers, the American Society of Mechanical Engineers, and the Institute of Electrical and Electronics Engineers. There were approximately 550 participants from 16 countries at the Symposium.

THE WHITE HOUSE
WASHINGTON

Santa Barbara

August 31, 1988

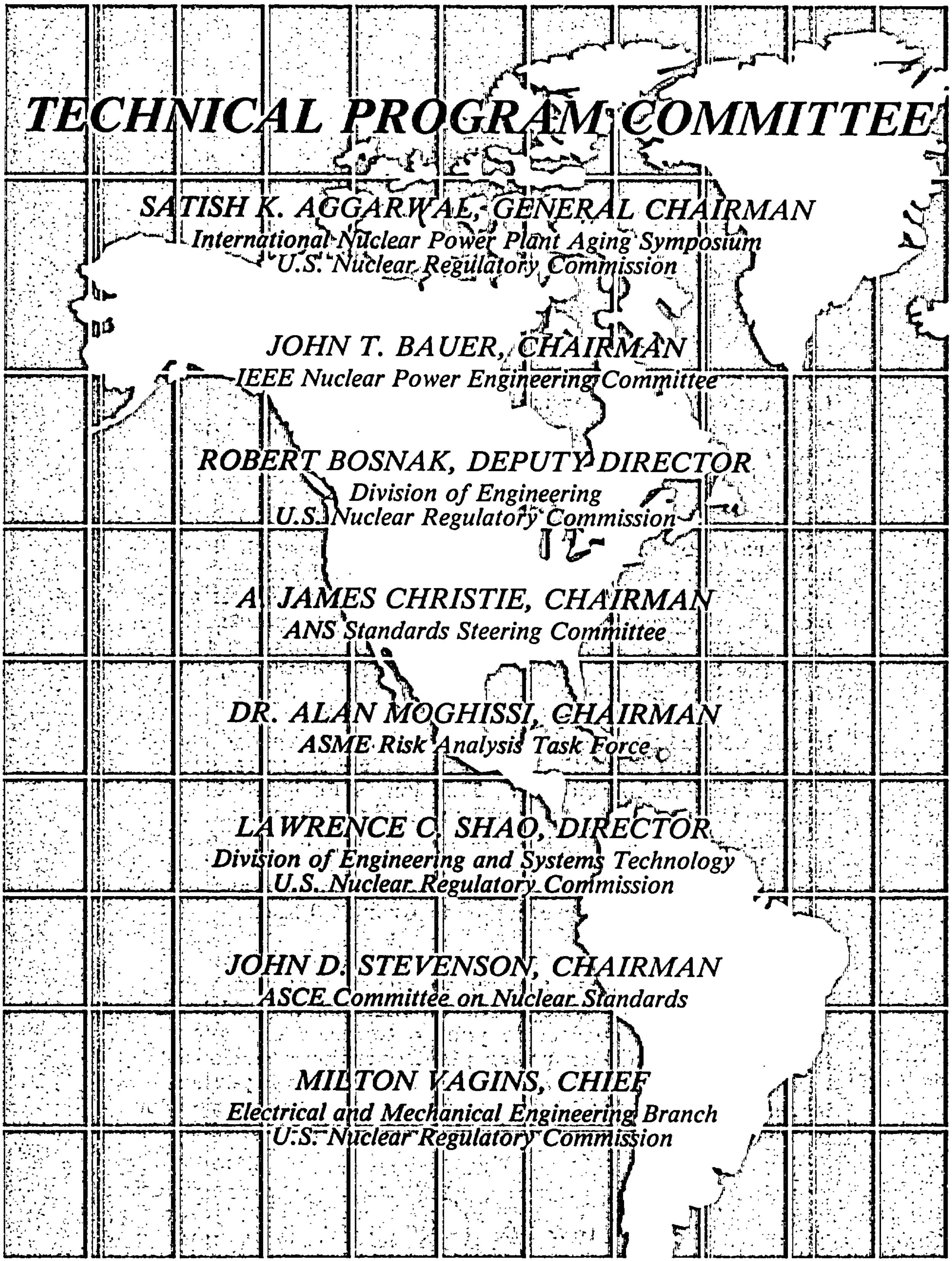
I am delighted to offer my warmest personal greetings to Chairman Satish Aggarwal and the many distinguished guests who have gathered from America and around the world for the International Symposium on Nuclear Plant Aging.

You are to be commended for your vigorous work in bringing your considerable engineering and scientific expertise to bear on the potential safety issues arising from the progressive aging of nuclear power plants. These issues must be addressed if our nations are to reap the benefits of a new age of safe, economical, and clean nuclear power.

It was with similar concerns in mind that Congress adopted, and I recently approved, a 15-year extension of the Price-Anderson Act. This law is designed to protect the public in the unlikely event of a nuclear accident and to make commercial nuclear power generation feasible. It also preserves the public's right to speedy compensation in the event of a nuclear accident and expands the level of protection to over $7 billion.

Meeting future demands for energy will require even more strenuous efforts to develop nuclear resources and to increase existing safeguards on their use. I wish all of you continued success in your determined endeavors to ensure safe nuclear power plant operations worldwide. With your strong commitment to resolve these important issues -- as indicated by your presence at this symposium -- I am confident you will reach your goal. God bless you.

Ronald Reagan



TECHNICAL PROGRAM COMMITTEE

SATISH K. AGGARWAL, GENERAL CHAIRMAN
International Nuclear Power Plant Aging Symposium
U.S. Nuclear Regulatory Commission

JOHN T. BAUER, CHAIRMAN
IEEE Nuclear Power Engineering Committee

ROBERT BOSNAK, DEPUTY DIRECTOR
Division of Engineering
U.S. Nuclear Regulatory Commission

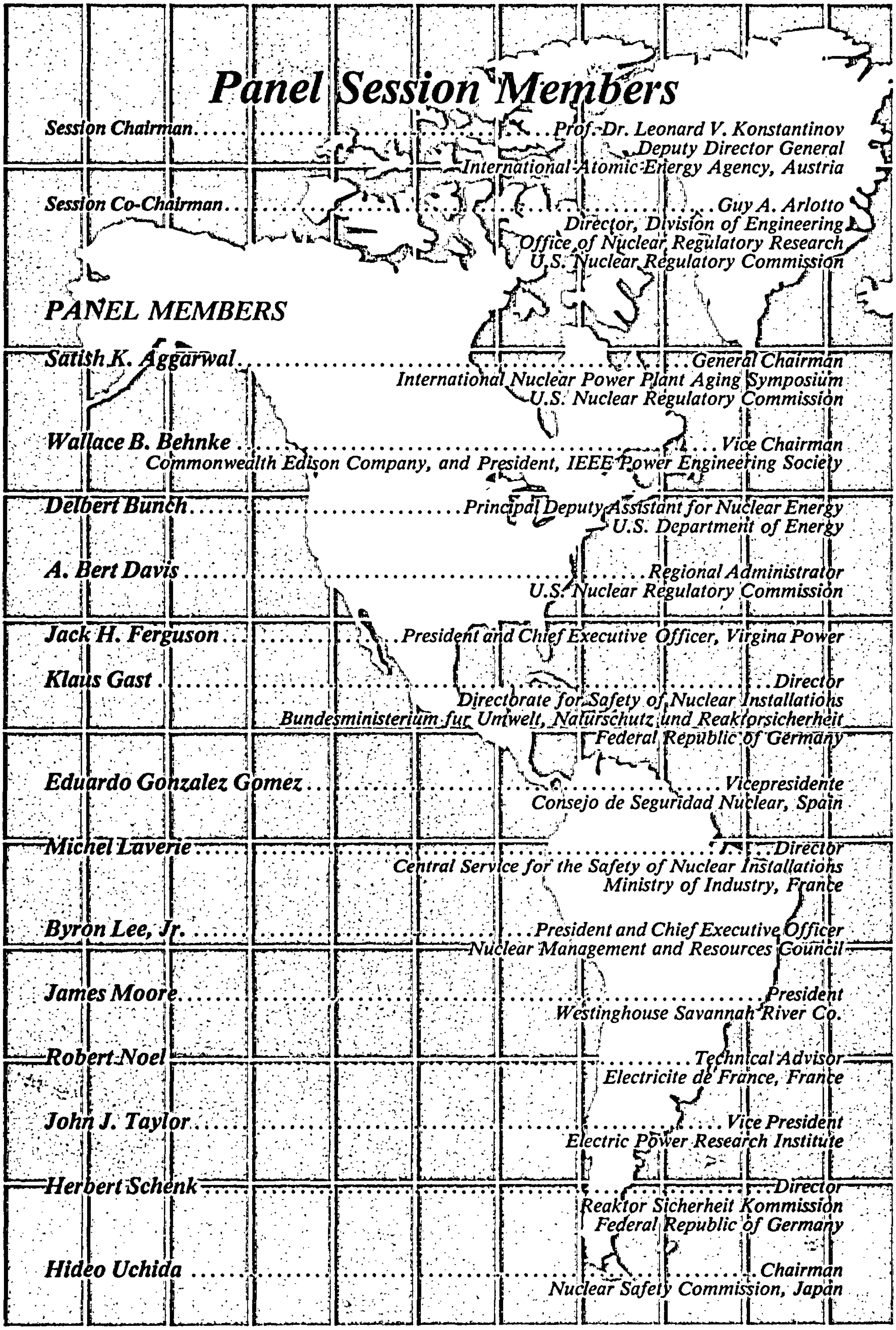
A. JAMES CHRISTIE, CHAIRMAN
ANS Standards Steering Committee

DR. ALAN MOGHISSI, CHAIRMAN
ASME Risk Analysis Task Force

LAWRENCE C. SHAO, DIRECTOR
Division of Engineering and Systems Technology
U.S. Nuclear Regulatory Commission

JOHN D. STEVENSON, CHAIRMAN
ASCE Committee on Nuclear Standards

MILTON VAGINS, CHIEF
Electrical and Mechanical Engineering Branch
U.S. Nuclear Regulatory Commission



Panel Session Members

Session Chairman	Prof. Dr. Leonard V. Konstantinov Deputy Director General International Atomic Energy Agency, Austria
Session Co-Chairman	Guy A. Arlotto Director, Division of Engineering Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission

PANEL MEMBERS

Satish K. Aggarwal	General Chairman International Nuclear Power Plant Aging Symposium U.S. Nuclear Regulatory Commission
Wallace B. Behnke	Vice Chairman Commonwealth Edison Company, and President, IEEE Power Engineering Society
Delbert Bunch	Principal Deputy Assistant for Nuclear Energy U.S. Department of Energy
A. Bert Davis	Regional Administrator U.S. Nuclear Regulatory Commission
Jack H. Ferguson	President and Chief Executive Officer, Virgina Power
Klaus Gast	Director Directorate for Safety of Nuclear Installations Bundesministerium fur Umwelt, Naturschutz und Reaktorsicherheit Federal Republic of Germany
Eduardo Gonzalez Gomez	Vicepresidente Consejo de Seguridad Nuclear, Spain
Michel Laverie	Director Central Service for the Safety of Nuclear Installations Ministry of Industry, France
Byron Lee, Jr.	President and Chief Executive Officer Nuclear Management and Resources Council
James Moore	President Westinghouse Savannah River Co.
Robert Noel	Technical Advisor Electricite de France, France
John J. Taylor	Vice President Electric Power Research Institute
Herbert Schenk	Director Reaktor Sicherheit Kommission Federal Republic of Germany
Hideo Uchida	Chairman Nuclear Safety Commission, Japan

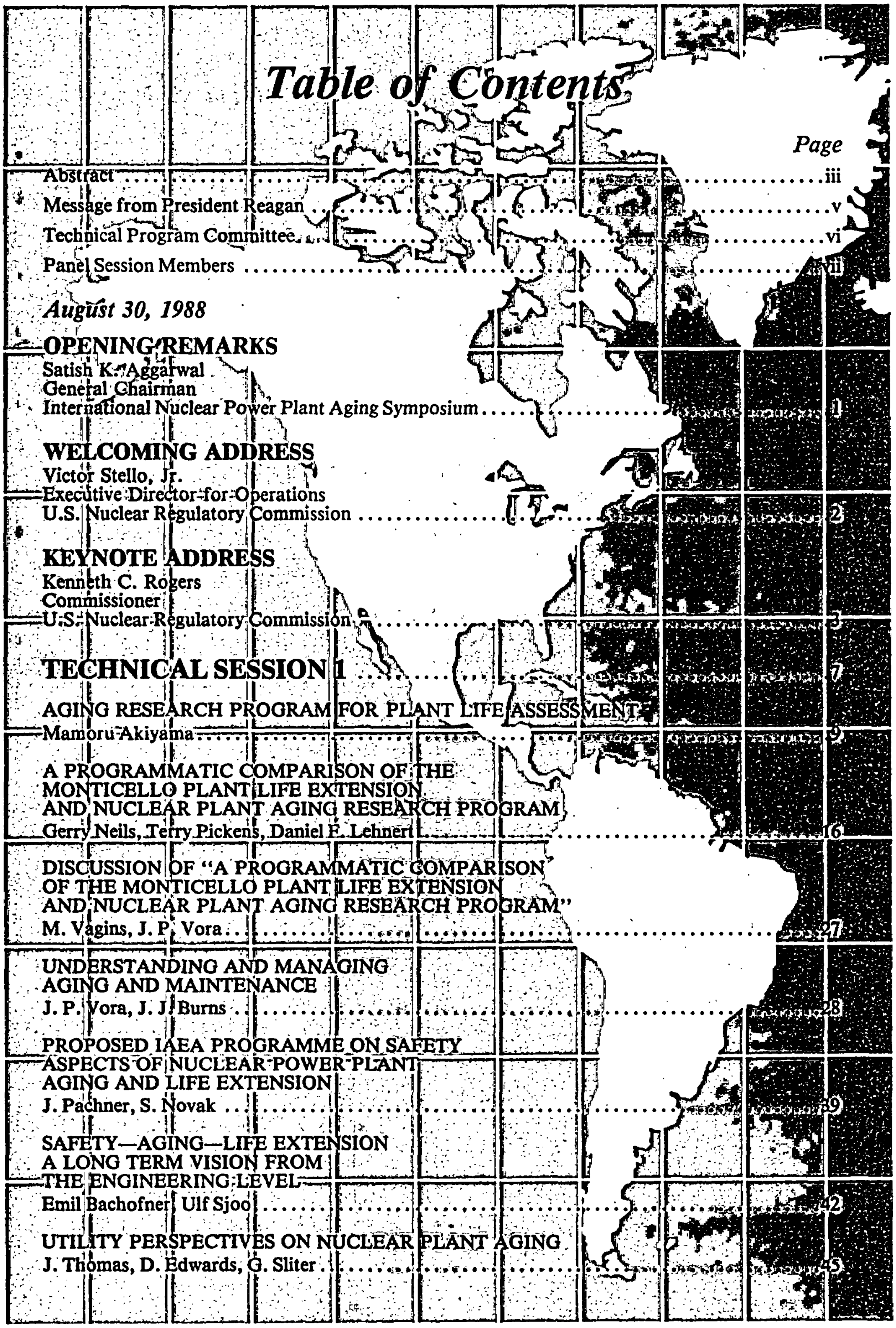


Table of Contents

August 30, 1988

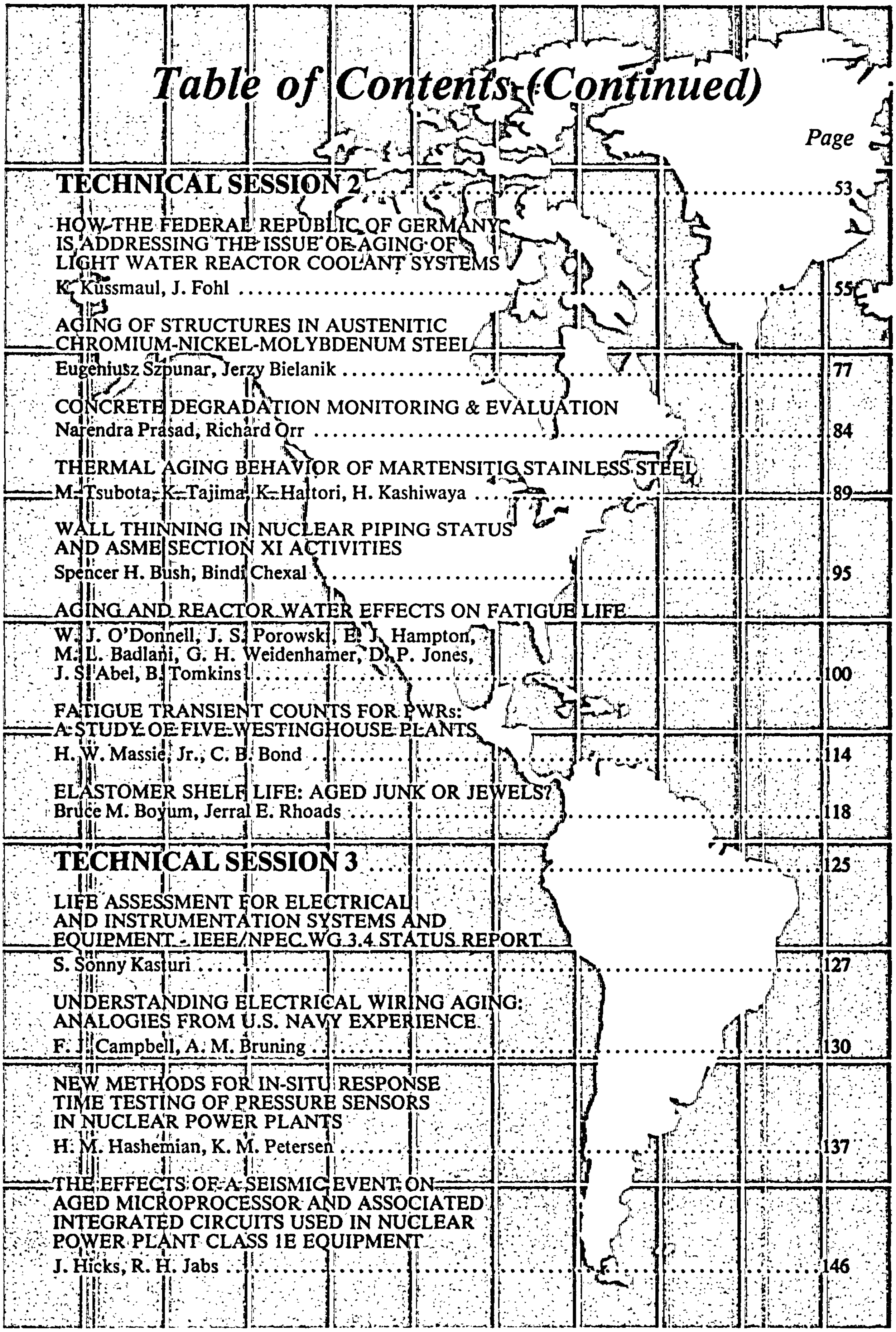


Table of Contents (Continued)

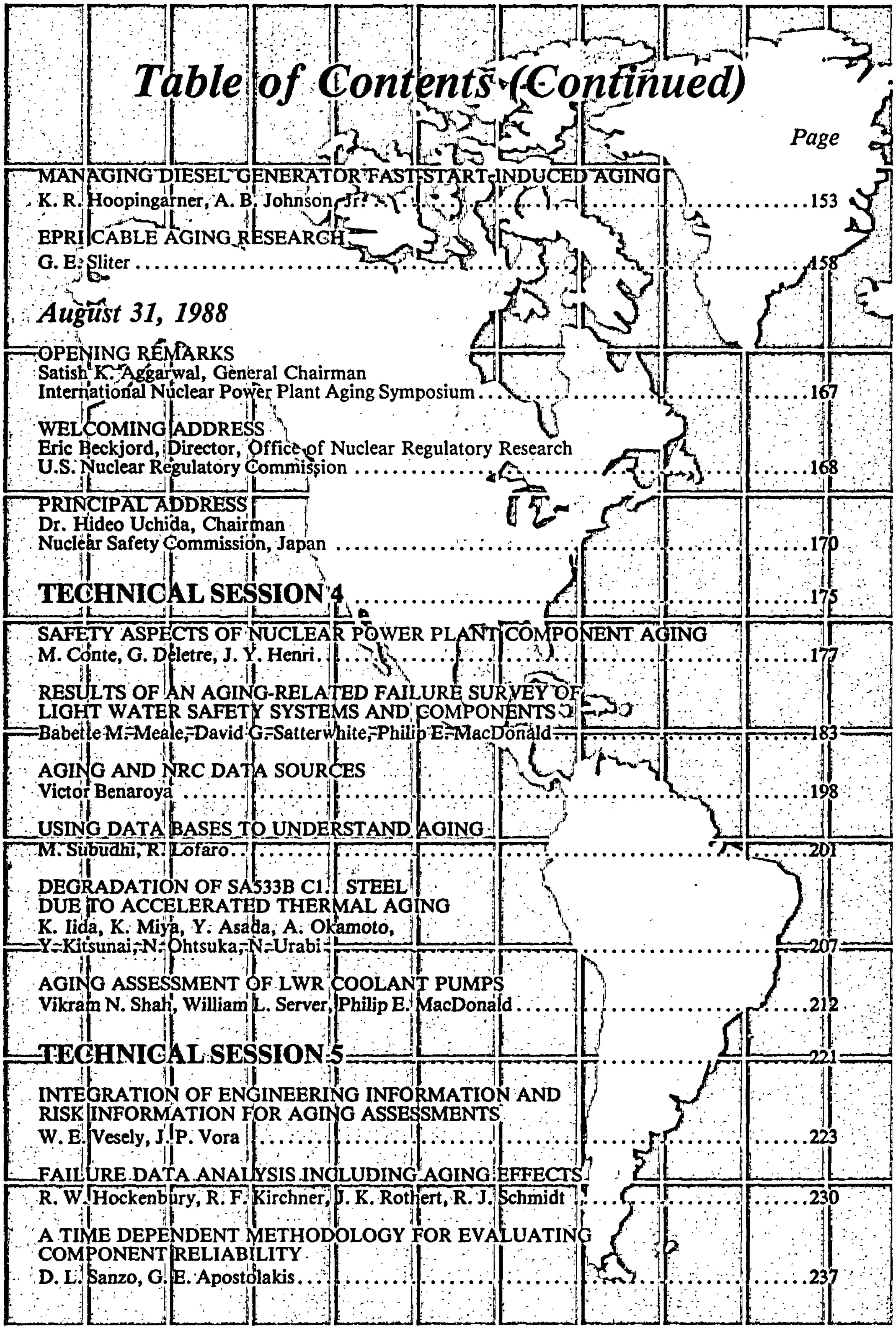
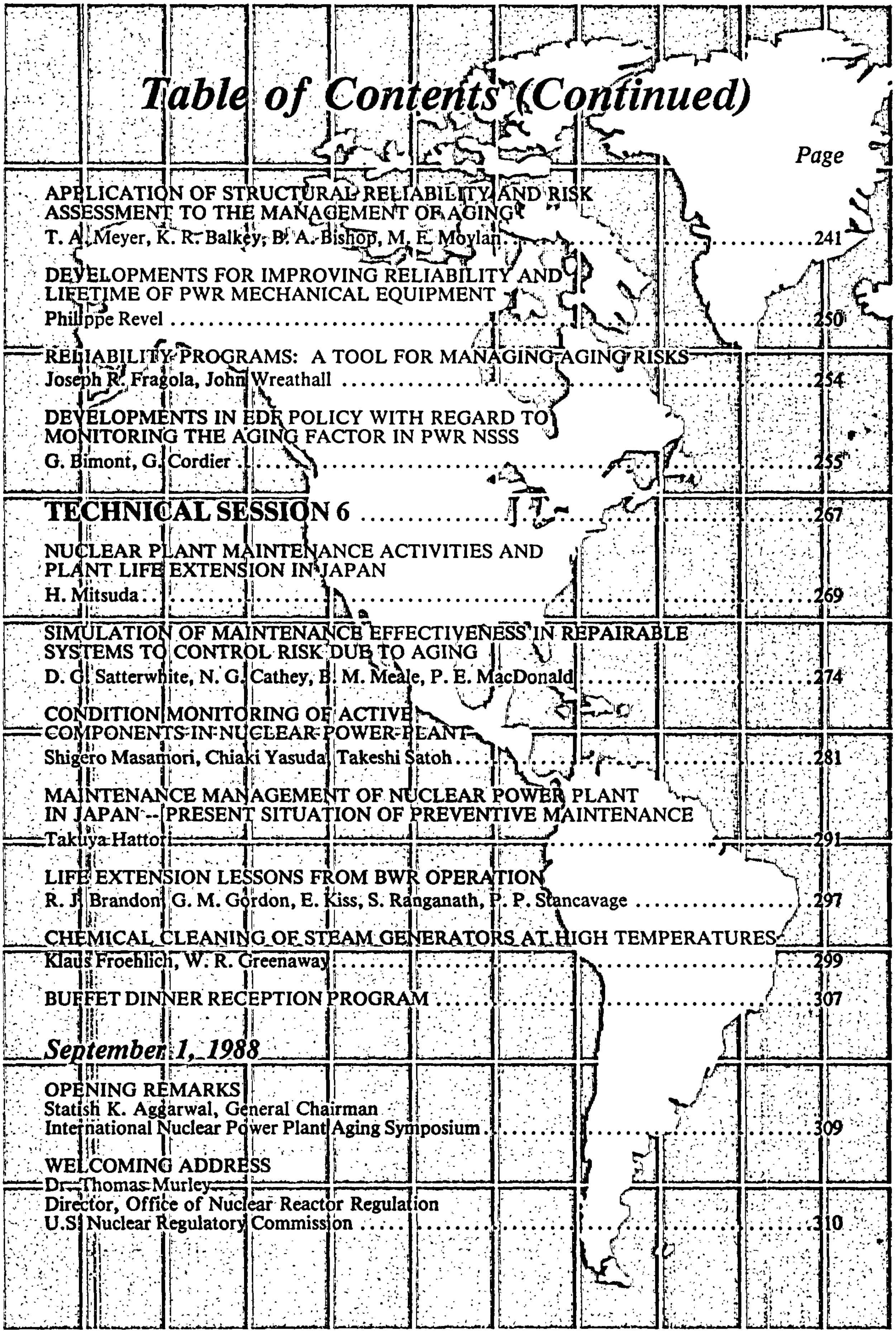


Table of Contents (Continued)

August 31, 1988

Table of Contents (Continued)



September 1, 1988

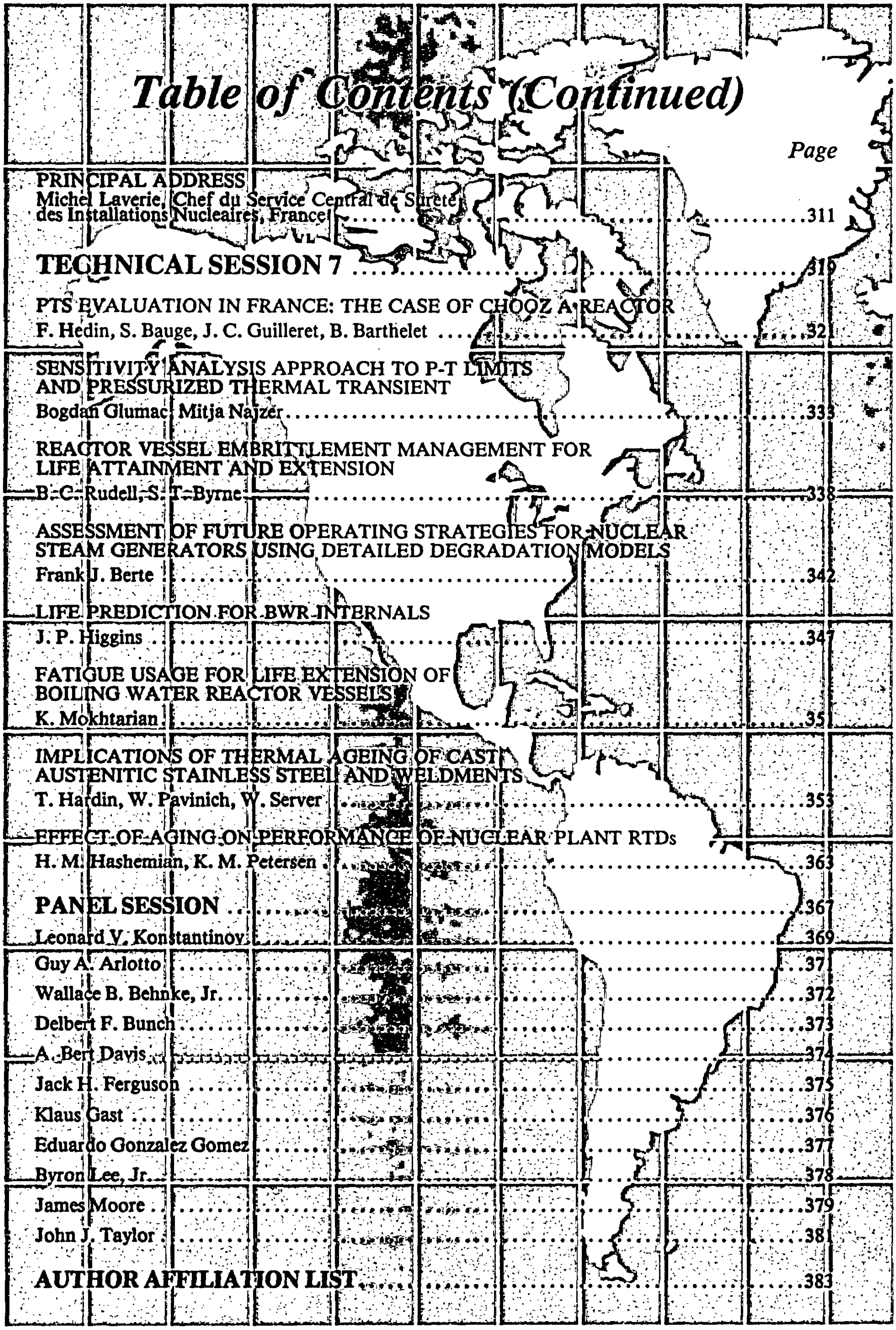


Table of Contents (Continued)

Page

August 30, 1988

OPENING REMARKS

Satish K. Aggarwal
General Chairman

As you all know, commercial nuclear power has been a reality for three decades. It has matured to the point that it plays a significant role in global electrical energy production and accounts for more than a quarter of the electricity generated in a number of countries.

With the passage of three decades of nuclear power plant operation, the international nuclear power community is now entering a period during which issues related to the progressive aging of nuclear power plants will play an ever-increasing role in the decision-making process for the continued safe operation of these plants.

In order for nuclear power to remain viable throughout the international community, it is absolutely essential that this option remain both safe and economical. Of these two essential characteristics, safety comes first and it is most crucial.

An accident anywhere in the world is of concern to all of us. We cannot ignore it. It has the potential for tragic human consequences. It threatens the nuclear option and its benefits everywhere in the world.

Safe and economical operations must continue in the future in spite of the fact that all is not yet known about the effects of aging on plant safety. No large nuclear power plant has yet entered its twenty-fifth year of operation, much less its fortieth. And yet, we must plan for safe operation not only for the present, but also for extended operation into the future.

But how do we formulate such planning in view of the many unknowns facing us? The answer, I believe, is right here in this room, and in fact is stated clearly in the theme of the Symposium, namely "Understanding aging, a key to ensuring safety; managing aging, a necessity to ensuring safety."

"Understanding" and "managing," these are what you and this Symposium are all about. You have embarked upon a rigorous international effort to understand plant aging. Your insightful research programs are directed to this understanding, as are your equally insightful operating data collection and mathematical analysis programs. From this understanding will flow the knowledge of how to safely manage aging through newly devised aging predictive techniques, and through new maintenance and corrective techniques rigorously devised to "maintain out" any and all potentially adverse effects of aging that can impact safe operations.

And from safe operation will flow considerable economic benefits, both for the present and for the future. Remember, safety comes first. Economy will follow from safety. It is not the other way around. We must all take the initiative worldwide to understand and manage aging. You must now ask yourselves what can we do to safely operate progressively aging nuclear power plants? I trust that the answer to this question will begin to unfold in the next three days in this Symposium.

WELCOMING ADDRESS

Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission

On behalf of the Nuclear Regulatory Commission, I'm certainly pleased to welcome you to the International Nuclear Power Plant Aging Symposium. I especially welcome our visitors from the many nations around the world who are present and participating in the Symposium. Their participation will give the Symposium a true international status, and that's as it should be, for plant aging and potential concerns related to plant aging recognize no national boundaries.

Nuclear power plants the world over are affected in many varied and not always understood ways by the aging phenomena. Thus, it is fitting that the world nuclear power community, acting together, should address this common issue in close cooperation. I see this Symposium as one important step in that process.

I am also gratified to note that progress has been made in this area over the past several years. There are several ways to measure this progress, and perhaps I am a little biased in my viewpoint as I mention our Workshop on Plant Aging held here in Bethesda in August 1982. At that time, as you may recall, we were beginning to appreciate the potential impact of plant aging phenomena on the continued safe operation of nuclear power plants. With this in mind, the NRC convened the workshop for the purpose of gathering ideas on how to best proceed to identify and resolve the various plant aging issues that we knew would be confronting us down through the years. In other words, we attempted to look into the future.

Now at this Symposium six years later, again here in Bethesda, you are reporting the results of your research studies, and your actual experience with plant aging phenomena. This Symposium is, of course, not the first forum in which such results have been presented, but for me, the Workshop and the Symposium serve to benchmark in a special way the progress that has been made.

In this regard I'm pleased to note the wide variety of technical matters that will be covered here. They are many and varied, and this is as it should be, for there are many plant aging issues, and they all need to be addressed fully.

Therefore, much remains to be done, and in the short time available to me this morning I urge you to develop, as I have, a deep appreciation for the full nature of the aging phenomena and the resulting need for increased international cooperation to resolve the potential safety concerns arising from these phenomena.

I am certain that a spectrum of recommendations for international cooperative efforts will be presented and discussed during the course of this Symposium and beyond. We will consider all such recommendations involving NRC very seriously, since it is only through a concerned and expanded effort involving all parties concerned that plant aging issues can be properly addressed and resolved.

Your agenda is a full one, and I know that during these three days much valuable information will be exchanged among you. Many new and creative approaches to the issues that confront us will be discussed, and many new initiatives will be developed and recommended.

I would like to add one more thought that the past few months brought to bear directly on plant aging, and that has to do with the Greenhouse Effect in the United States. The concern about use of fossil fuels has resulted in proposals, including legislation, for expanded use of nuclear power. Extending the life of current operating plants beyond 40 years will receive increased priority. Therefore, there is an added and important incentive to understand aging issues, not just for current operations, but also for resolving issues related to plant life extension.

So with that brief introduction, let me wish you very, very much success in the Symposium for the next few days.

KEYNOTE ADDRESS

Kenneth C. Rogers
Commissioner
U.S. Nuclear Regulatory Commission

On behalf of the U.S. Nuclear Regulatory Commission, I am delighted to welcome you. I am particularly pleased to see the fine international attendance at this Symposium on what, of course, is an important and universal problem.

Chairman Zech would have liked to be here himself to initiate these proceedings, but he is travelling overseas at this time. He has asked me to convey to you his regrets at not being able to be here and his best wishes for the success of this meeting.

Before I address the main theme of the Symposium, I would like to recognize and thank the co-sponsors of this Symposium. They are the American Nuclear Society, the American Society of Civil Engineers, the American Society of Mechanical Engineers, and the Institute of Electrical and Electronics Engineers. Without their support, this Symposium would simply not have been possible.

I will speak to you briefly this morning of my perspectives on the aging problem in nuclear plants. These will be primarily the perspectives of a Commissioner of a United States regulatory agency, but also those of an experimental physicist who has had extensive training and experience with the behavior of a wide variety of materials and systems, and who understands, I think, some of the generic issues associated with aging.

It has been my observation since coming to the NRC that the aging of nuclear power plants is one of the most important issues facing the nuclear industry worldwide and the governmental bodies which provide safety oversight and regulation. As evidence, one need only cite an example such as the main feedwater pipe break at Surry Unit 2 in December 1986 in which there were four fatalities. The post-accident investigation showed that the pipe failed because of a thinned wall, which probably resulted from a corrosion/erosion mechanism, one form of aging. These conclusions were rapidly communicated to the entire world-wide nuclear community.

While this incident is particularly significant because of the loss of life involved, nuclear power plants have over the years experienced a number of component degradations and even failures resulting from a variety of aging processes.

Aging encompasses all forms of degradation to nuclear power plant materials, components, and systems that result from exposure to environmental conditions in the power plant or from operational characteristics or procedures. The agents of aging range from radiation, high temperatures and pressures, steam conditions, and corrosive chemicals in the power plant environment; to thermal and pressure cycling, component vibration, and mechanical wear of operation; to the degradation induced by inspection processes or other such procedures.

Virtually every component in a reactor is subject to some form of aging, from the pressure vessel itself to the fuel and cladding, the pipes, the pumps and valves, and the electrical systems that make up the rest of the plant.

Aside from the corrosion/erosion experienced by the Surry plant, aging phenomena can, depending on the component and the conditions, include embrittlement, fatigue, corrosion, cracking, mechanical wear, creep, and dimensional instability.

While failures of individual components constitute an operational concern and can be a safety concern, the more significant safety concern results not so much at a single-component level, but at the higher level of component aggregation, because our key safety systems have been designed to accommodate single failures.

We have been particularly concerned about common-mode failures. For example, several forms of aging degradation of steam generator tubing have been identified. The concern is not a single tube leaking or even failing, the concern is with sudden multiple tube failures, common-mode failures. For example, such failures could come about by having essentially uniform degradation of the tubes.

Degradation could decrease the safety margins so as to create a "loaded gun," an accident waiting to happen. Under those conditions, a pressure transient or a seismic event could rupture many tubes simultaneously. That could allow primary coolant to enter the secondary system and the resulting high pressure to lift the relief valves that are outside containment on the steam line, thus permitting primary water to bypass containment and communicate with the atmosphere directly, resulting in a loss-of-coolant accident.

Another concern is loss of defense in depth. A basic tenet of reactor safety design has been the incorporation of redundant and diverse systems to prevent and mitigate accidents. For example, the reactor protection system, the emergency core cooling system, the containment spray system, and the on-site power system are all designed to assure multiple options for mitigation of a breach in the reactor pressure boundary. However, if this engineered redundancy and diversity is gradually and unknowingly reduced because of aging, then safety margins will eventually be seriously reduced.

Plant aging is particularly important in the United States, which has an older plant inventory, on the average, than does the rest of the world. Whereas nearly two-thirds of the power plants in the rest of the world are under ten years old, in the United States about two-thirds are over ten years old. Of course, the United States is certainly not alone in facing aging problems. Great Britain, for example, still operates reactors dating from the mid to late 1950s, and a number of countries have plants that are 20 or more years old.

In the United States, we can also anticipate that the problems of aging power plants will be exacerbated by owners' desires to continue to operate existing reactors beyond the somewhat arbitrary 40-year period for which most of them are presently licensed. This problem is also not unique to the United States, but is likely to be particularly important here because of the very small amount of new electrical power production capacity of any type that has been installed in recent years.

Given the long lead times and the high costs typically required to bring any new plant on line, the continued operation of existing plants may be very important to meeting growing electrical power demands.

Thus, we can anticipate that while we first and foremost need to be concerned about the continued safe performance of aging nuclear power plants in the near term, we will also likely have to address the question of how much longer than 40 years they can operate safely, and what is required to ensure the safety of such operation. Other countries have different reactor licensing requirements and may or may not have to address the question of extending a license for a fixed period of time.

Fortunately, there are a variety of measures that can be taken to "manage" aging. By managing aging, I mean predicting or detecting when a component or system has degraded to the point that it becomes a potential safety hazard and taking appropriate corrective measures.

There are a variety of detection techniques and remedial actions that I am sure are familiar to all of you, so I'll merely mention them here.

Detection can include continuous or periodic monitoring (for example, of acoustic emissions for valve leak detection, or of water chemistry to minimize the potential for corrosion, corrosion/erosion, and intergranular stress corrosion cracking), visual and nonvisual inspection of components either during service or during outages (for example, eddy current testing of steam generator tubes or ultrasonic tests for crack detection in pipes, welds, and flanges), and periodic functional testing of equipment (such as cycling valves open and closed in standby safety systems). Remedial measures can include component replacements or such techniques as in situ annealing of reactor pressure vessels to restore required mechanical properties.

For all such management measures, questions that must be addressed are how much to inspect or test, how often to repair or replace, and what constitute signals of degradation that should alert us. These are very important and basic questions and are at the center of what you will be discussing this week.

I want to emphasize as strongly as I can that aging problems cannot be addressed simply by building in what has been called conservatism. In my view, our conservatism has not always been conservative. When we stitch one margin of safety on top of others for no sound technical reason but only in the hope of ensuring that we have covered any possible but unanalyzed error or unknown, we often inadvertently establish a situation that is not conservative at all, and we don't even realize it.

Because some testing is good, more is not necessarily better. In fact, more is sometimes less safe because inspection and testing are not necessarily inherently benign. Tests of equipment may themselves put strains on the equipment. We have discovered this, as we all know, in the case of diesel generators, where requirements to test the equipment by rapid starting from a cold condition have been found to put significant wear and tear on the generators. Thus, testing may in fact reduce safety if it makes a generator fail just when it is needed.

Similarly, inspections may introduce wear on equipment or increase the probability for damage or errors in reassembling or reinstalling if disassembly had been required for the inspection.

Another aspect of the management of aging is the need for a knowledge base of plant history in establishing residual safety margins for components and systems. Knowledge of the numbers of cycles that components and systems have experienced, the maximum temperatures and pressures they have experienced, the total number of hours of operation, and the cumulative exposure to corrosive chemical elements can all be related directly to the amount of aging that components and systems have experienced.

As the inventory of nuclear plants gets older and certain systems and components approach regulatory limits for operation, the need for good records to establish the plant history will increase. I can see this activity becoming part of the integrated maintenance data bases many utilities are now developing.

Finally, I note the importance of taking a systems approach to the challenge of managing aging. That is, aging cannot be addressed in isolation. Aging and maintenance share a symbiotic relationship. Maintenance requirements are, of course, significantly shaped by needs to counteract the effects of aging. Conversely, maintenance procedures can produce aging.

I have touched on how accident problems can be exacerbated by or even produced by aging. Aging clearly has an impact on the qualification of electrical equipment. Certain human actions may also be important to aging. For example, human error may produce additional stresses on the system and thereby accelerate aging.

Thus, we cannot think that we are addressing the aging problem simply by looking at environmental interactions with materials and components. Rather, we must consider both the impacts of various activities on aging and the impacts of aging on the entire system.

While we have, based on past experience, identified a number of aging problems and developed appropriate ways of managing them, I don't believe we know yet all we need to know about these problems. Operational experience continues to produce evidence of situations we did not anticipate. Research on known problems has demonstrated that some of our initial assumptions were either incorrect or incomplete. Thus, I believe that a very important element of aging management is a strong, continuing research program.

Research yields several important benefits. It can, of course, be used to examine clearly identified problems and to develop appropriate ways of managing them. Such measures can include the development of advanced monitoring and repair techniques, and, where appropriate, the technical bases for the reduction of unnecessarily restrictive management measures that may diminish safety.

For example, better understanding of the relationship between physical measures of degradation and resulting impacts on reliability and performance, and better ways of predicting rates of degradation over time can allow one to set more precise requirements on the scope and frequency of inspection and maintenance.

Research can also be used to reveal potential new sources of degradation, not yet observed in practice, before they become problems in existing plants. Accelerated aging experiments, controlled tests, and the development of computer models to explore material and component performance in domains beyond the design basis can enhance the understanding of the behavior, during abnormal conditions, of components with various degrees of aging.

Research can help develop a better understanding of other important safety concerns. For example, our analyses of severe accidents have to date suffered in that the probabilistic risk assessments made have not in general been able to recognize the effects of aging on the probabilities of certain kinds of events. Incorporating aging effects into PRAs will give deeper insight into the understanding of severe accidents and help us to address that problem.

Finally, research can provide additional capabilities to answer new questions rapidly and effectively as they arise. Several times in the past, NRC has been able to resolve significant new safety issues rapidly because it had an accumulated body of high quality research data and available experts to examine the problem.

One such case was the pressurized thermal shock (PTS) problem, which involved the question of whether overcooling transients occurring in conjunction with high coolant pressure could result in stresses causing pre-existing cracks to propagate through the pressure vessel wall. Because NRC had conducted research for 15 years on vessel aging, it had data available on reactor vessel steel aging, together with crack propagation analyses that already had been validated by the age-related data. These together showed that even the most susceptible vessels had a safety margin sufficient to permit generic resolution through rulemaking without plant shutdowns.

I'll leave it to Mr. Beckjord and others speaking at this conference to describe NRC's nuclear plant aging research program in detail. I think our program is a strong one, and it is making significant contributions to the management of aging and to the improvement of plant safety.

The plant aging research program's broad scope has allowed it to address numerous aging problems requiring different expertise and diverse research facilities. The long-term and continuing investment NRC has made in this program has given us the background and tools to answer questions that were not even anticipated when the research was initiated or performed.

This breadth and stability must be maintained in the future. We can do so, in part, by continuing our commitment to cooperative research programs with U.S. industry and with foreign countries and by carefully selecting areas for research with a long-term perspective in mind.

Domestic and international cooperative research efforts have proven one of the most important cost-effective and productive elements of our research program. They provide opportunities for the synergism that can come from a broad range of views and approaches, and they effectively extend the investments of each research partner through the sharing of expensive facilities, the division of labor in experimentation and analyses, the elimination of duplicating work already done, and the sharing of information.

This Symposium, of course, is an excellent manifestation of international collaboration. Other efforts I could cite include the recently completed five-year cooperative program on steam generator integrity with Japan, Italy, France, and the U.S. industry-sponsored Electric Power Research Institute (EPRI); a new three-year cooperative program on piping integrity (the IPIRG program), which includes France, the United Kingdom, Canada, Sweden, Switzerland, Japan, Taiwan, and EPRI; a cooperative program with Germany and the United Kingdom for testing and evaluation of pressure vessel steel from the German Gundremmingen reactor; participation with 14 countries in the international cyclic crack growth rate cooperative program to study the environmental effects of reactors on cracking in steels used in light-water reactors; and membership in the newly formed international cooperative program on irradiation-assisted stress corrosion cracking of LWR core internals. Every one of these cooperative programs falls under the "aging" umbrella.

The appropriate selection of research areas is itself an important activity. First, none of us have sufficient resources, facilities, or manpower to address in detail every possible combination of aging phenomena and susceptible components. Therefore, we must focus our efforts on those items that have the greatest potential to impact safety. We must, of course, be assured that the selection of priority research areas covers a broad range of types of components and types of problems so that, should a new problem be identified, we're likely to have done research in at least a related area.

Based on these considerations, categories of components that we believe have a potential for aging-related impact on plant safety include mechanical components such as motor-operated, check, and solenoid-operated valves, power-operated relief valves, snubbers, compressors, heat exchangers, and pumps; electrical components such as cables, circuit breakers, relays, chargers, inverters, batteries, motors, bistables, transformers, connectors, and electric penetrations; and systems such as the high- and low-pressure emergency core cooling systems, the residual heat removal systems, and the auxiliary feedwater systems.

A key requirement in recent years has been the need to take advantage of special time-critical opportunities to obtain actual reactor samples. For example, we have put considerable priority on obtaining exposed samples and components from facilities being shut down or decommissioned, such as the Shippingport Nuclear Power Plant in Pennsylvania. As such reactors are entombed, dismantled, and components disposed of, a unique opportunity to study their aged components appears and then disappears.

I'd like to leave you with some thought as to how I believe an aging program might evolve in the future. Perhaps you can consider these during the course of the Symposium and in forward planning within your own institutions.

We need to take fuller advantage of relevant experience from other industries. While there are, of course, some unique aspects to nuclear power plants, they still share many components, materials, and environmental conditions with petrochemical plants, fossil fueled plants, aircraft, ships, and other complex systems. We should make maximum use of the experience of these industries where it is applicable. The concept of using data from other industries is not new. We have done so before, for example, in acquiring data for probabilistic risk assessments. I believe joint research and coordination with other industries in selected areas could help to ensure a mutual exchange of relevant information.

We need to be more creative and far-sighted in developing advanced instrumentation and control systems to monitor and inspect nuclear plant components. I believe there is much more to be done in this area.

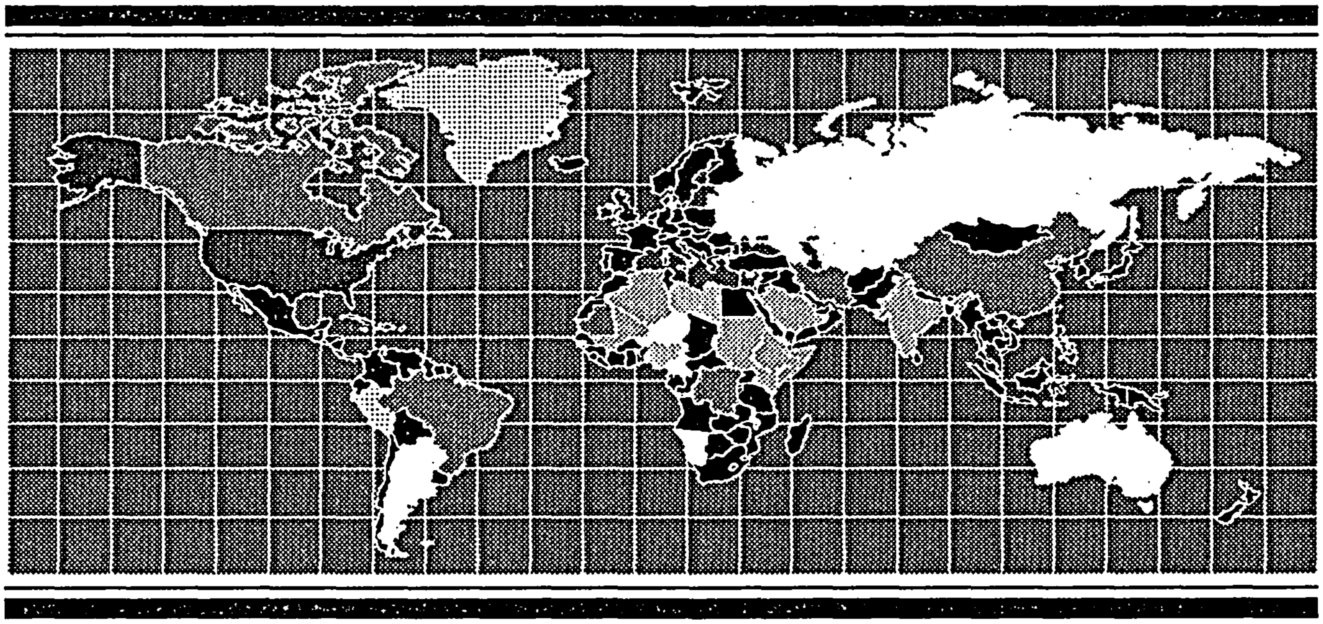
We need to understand better the interrelationships between aging and other concerns to be able to implement the systems approach I spoke of earlier. I have already mentioned the interfaces of aging and severe accidents, maintenance, human factors, and equipment qualification. I also see a need to apply the experience we have gained on the aging of present reactor components to the design of the next generation of reactors.

In concluding, I would like to say that if one thing impresses me more than anything else about the aging problem, it is the numerous linkages it has to other industries, to other areas the NRC is addressing, to other domestic organizations, and to other countries. I have tried to point these out. These linkages make study of the aging problem more difficult in some respects, but they also provide great opportunity. I think these linkages will shape our efforts to manage aging better and better in the years ahead.

Aging is clearly a very important area of concern, and it will only grow more important as reactors get older, and as we face the question of operating reactors beyond the 40 years presently authorized in the United States. Not only can we expect to continue to experience those aging problems we have already seen and studied, but we will most assuredly discover new aging effects at levels of exposure and conditions beyond those we have yet encountered.

To assure continued safety under these conditions, we will need information from all relevant sources, continued strong research efforts, and as much international commitment and cooperation as possible.

Thus, I commend you all and I hope your next few days are productive and beneficial for the advancement of world-wide nuclear plant safety.



TECHNICAL SESSION 1

Aging Research Programs

August 30, 1988

Session Chairman

DR. DENWOOD F. ROSS
Deputy Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Session Co-Chairman

DR. GAIL H. MARCUS
Technical Assistant to Commissioner Rogers
U.S. Nuclear Regulatory Commission

AGING RESEARCH PROGRAM FOR PLANT LIFE ASSESSMENT

MAMORU AKIYAMA

Professor, University of Tokyo
Chairman, MITI Technical Advisory Committee*
JAPEIC PLEX Committee** Member

SUMMARY

Ministry of International Trade and Industry (MITI) has judged it is very important for nuclear power plants in Japan to reduce the electric power generation cost by means of development of plant life extension technology.

MITI started a 8-year program of "Development of Nuclear Power Plant Life Extension Technology" from 1985, and has entrusted this program to Japan Power Engineering and Inspection Corporation (JAPEIC).

JAPEIC has established the nuclear plant life extension technical committee consisting of experts from universities, utilities and manufacturers and has been executing the program under the direction of the committee.

This paper describes the results of preliminary life assessment on major components/structures of nuclear power plant and the aging degradation researches selected on the basis of the preliminaty life assessment, which are being conducted in this project covering the following items, and our whole plant life extnsion technology development program is also introduced.

1) Evaluation of corrosion fatigue degradation of low alloy steel and austenitic stainless steel materials used for reactor pressure vessels and primary coolant piping systems.
2) Evaluation of thermal aging degradation of duplex stainless steel castings used for PWR primary coolant pumps and primary coolant piping systems.
3) Evaluation of irradiation SCC of austenitic stainless steel used for BWR internals.
4) Evaluation of irradiation embrittlement and fatigue of austenitic stainless steel used for PWR inernals.

We blieve that these researches will contribute to not only life assessment of nuclear power plants, but also to assessment of plant reliability.

1. Introduction

The nuclear power generation by light water reactors in Japan has accounted for 29% of the total power generation(185×10^6MWh, as of the end of 1987 calendar year), and it is expected that these LWR plants will continuously play the great role in power supply capability for a long time.

In view of this situation, the Ministry of International Trade and Industry (MITI) has formulated the Light Water Reactor Technology Sophistication Program which is designed to enhance the technology for existing light water reactors, to develop advanced light water reactors and to investigate next-generation LWR concepts, thereby further increasing safety, reliability and economy of light water reactors. This Sophistication Program is being translated into action step by step.

The Nuclear Power Plant Life Extension Program is a part of the program for enhancement of existing light water reactor technology, implemented of a 8-year program starting from fiscal 1985. The objective of this program is to develop plant life extension technologies such as those for prediction of plant life and replacement/repair of critical components/systems to extend the design service life of plants. As a result of the program, the amount of the total-life power generation will be enlarged and the life-averaged power generaiton cost will be reduced.

The Japan Power Engineering and Inspection Corporation (JAPEIC), entrusted by MITI with the research and development work for this program, organized the "Nuclear Power Plant Life Extension Technical Committee" with scientists and experts from universities, electric utilities and manufacturers.

This paper describes the results of preliminary life assessment on major components/structures of nuclear power plant and the aging degradation researches selected on the basis of the preliminary life assessment, which are being conducted in this project covering the following items, and our whole plant life extentsion technology development program is also introduced.

2. Preliminary Life Assessment Results

Preliminary life assessment has been conducted on major components and concrete structures, which are likely to govern the nuclear power plant life, by assuming domestic 800 MWe class BWR and PWR plants, and necessary aging degradation data for the assessment have been identified, to select the aging degradation research items which will be implemented in this project.

• Technical Advisory Committee on Nuclear Power Generation to the Minister of International Trade and Industry

•• Japan Power Engineering and Inspection Corportion Plant Life Extension Technical Committee

(1) Major Components

The number of major components which have been selected and subjected to the preliminary life assessment was 10 for BWR plant and 13 for PWR plant. In selecting these components, the feasibility of replacement and repair of the components, the effect of loss of function of the components on plant, amount of exposure, etc., were taken into consideration. In conducting the preliminary life assessment of major components, each components were broken down to subcomponents, and lives of these subcomponents were evaluated for various aging degradation causes based on the currently available technology. Examples of results of such preliminary life assessment conducted on PWR reactor pressure vessel, BWR reactor internals and PWR primary coolant piping are presented in Table 1, 2 and 3.

In these tables, items marked "○" are factors governing aging degradation which have large effects, those marked "△" are factors governing aging degradation but having small effects, and those maked "※" are factors which may affect aging degradation, but difficult to be quantified at this stage. The preliminary life assessment have been conducted on factors categorized "○" and "△" by assuming that these factors do not act simultaneously.

The life assessments were conducted based on the fatigue evaluation method given in ASME Section Ⅲ for the fatigue degradation, on the embrittlement prediction methods given in NRC RG 1.99 and ASME Section XI for the irradiation embrittlement, and on the existing design data for the general corrosion.

For PWR reactor pressure vessel, fatigue, general corrosion, and thermal aging are relevant aging degradations on most of its subcomponents, as indicated in Table 1, and irradiation embrittlement is relevant for the core shell. The results of preliminary life assessment concerning fatigue degradation, irradiation embrittlement and general corrosion indicated that PWR pressure vessel has lives of 100 years or more except for core shell and stud bolts, but it has been pointed out that fatigue data of low alloy steels in corrosion environment, including thermal aging effects, are required for quantitative evaluation. A verification testing has been conducted for core shell irradiation embrittlement.

For BWR reactor internals, the aging degradations caused by fatigue, SCC, irradiation embrittlement, and thermal aging will be issues. Preliminary assessment indicated that they have lives of more than 100 years against fatigue, however, it was pointed out that aging data on stainless steels are required concerning irradiation SCC, irradiation embrittlement and thermal aging.

For PWR primary coolant piping, fatigue and thermal aging are controlling factors. Although related subcomponents have lives of more than 70 years except for surge line nozzle, it was found out that thermal aging data of duplex stainless steel are required.

TABLE 1 PWR Reactor Pressure Vessel Preliminary Assessment

No.	Subcomponents	Material	Aging Degradation Factor							Note) Preliminary Life Assessment
			Fatigue	SCC	Irradiation Embrittlement	Thermal aging	General Corrosion	Abrasion	Others	
1.	Shell (core part)	SA508Cl.2	○		○ (※)	※	△			40y. or more
2.	Nozzle (feedwater)	SA508Cl.2	○			※	△			100y. or more
3.	Stud Bolt	SA540B24Cl.3	○				△			50y. (replaceable)
4.	Other Subcomponents	Alloy 600	○	※						100y. or more
5.		SA508Cl.2	○			※	△			
6.		SA533Gr.BCl.1	○			※	△			

○ : Degradation factor whose effect is large.
△ : Degradation factor whose effect is small.
※ : Factor which may affect, but can not be quantified now.
Note : Preliminary assessment for either ○ or △. (Reassessment is requited for ※.)

TABLE 2 BWR Reactor Internals Preliminary Life Assessment

No.	Subcomponents	Material	Aging Degradation Factor							Note) Preliminary Life Assessment
			Fatigue	SCC	Irradiation Embrittlement	Thermal aging	General Corrosion	Abrasion	Others	
1.	Top Guide	Type 316 SS	△	※	※	※				100y. or more
2.	Shroud	Type 304 SS	△	※	※	※				100y. or more
3.	Core Support Plate	Type 304 SS	△	※	※	※				100y. or more
4.	Other Subcomponent	Type 304 SS	△	※		※				100y. or more

○ : Degradation factor whose effect is large.
△ : Degradation factor whose effect is small.
※ : Factor which may affect, but can not be quantified now.
Note : Preliminary assessment for either ○ or △. (Reassessment is required for ※.)

TABLE 3 PWR Primary Coolant Piping Preliminary Life Assessment

No.	Subcomponents	Material	Aging Degradation Factor							Note) Preliminary Life Assessment
			Fatigue	SCC	Irradiation Embrittlement	Thermal aging	General Corrosion	Abrasion	Others	
1.	Pipes	SA351 CF8M	○			※				100y. or more
2.	Surge Line Nozzle	SA182 F316	○							56y.
3.	Residual Heat Removal System and Safety Injection Nozzle	SA182 F316	○							71y.
4.	Other Subcomponents	SA182 F316	○							100y. or more

○ : Degradation factor whose effect is large.
△ : Degradation factor whose effect is small.
※ : Factor which may affect, but can not be quantified now.
Note : Preliminary assessment for either ○ or △. (Reassessment is required for ※.)

(2) Preliminary Life Assessment of Reinforced Concrete Structures

Concerning reinforced concrete structures of nuclear power plnat, the preliminary life assessment was conducted based on existing methodology on base mat, reactor pedestal, primary shield wall, steam generator wall and intake structure.

The salt content, carbonation, alkali-aggregate reaction, heat and radiation were considered as factors causing aging degradation, and the thickness of concrete cover, water/cement ratio, temperature, humidity, carbon dioxide concentration and chlorine concentration were taken as parameters.

The assessment indicated that all reinforced concrete structures of nuclear power plants, in general, have lives of more than 80 years.

However, it is deemed necessary that examination of aged concrete properties must be conducted, including presence of cracks, and it is planned to conduct core sample tests on the structures of old plants.

3. Aging Degradation Research Program

Possible causes of aging degradarion causing in components/structures of nuclear power plant are thermal aging, neutron irradiation, corrosion fatigue, stress corrosion, general corrsion, fatigue, abrasion, etc., and each of these factors, or combination of them, may create such degradation as reduction of fracture toughness (embrittlement) , crack propagation, reduction of thickness, etc..

However, there is little data which can be used for quantitative evaluation of such aging degradation phenomena, and the appropriate methods seem to be unavailable for predicting aging degradation at present time. For this reason, the following four items of verification tests in this technology development project has been planned for the objective of collecting the aging degradation data required for life evaluations which are identified in the preliminary life assessment described above and developing aging degradation prediction methodology.

* Evaluation of corrosion fatigue degradation for reactor pressure vessels, primary coolant piping sysems, etc.
* Evaluation of thermal aging degradation of duplex stainless steel for PWR primary coolant piping systms and pumps.
* Evaluation of irradiation SCC for BWR reactor internals.
* Evaluation of irradiation embrittlement for PWR reactor internals.

The implementation plan for conducting these evaluations are described in the following.

In reactor pressure vessel, the embrittlement by neutron irradiation is also an issue. The predictive equations for irradiation embrittlement are now being developed in another project in JAPEIC (Reactor Vessel Pressurized Thermal Shock Verification Test). Certain aspects of this development are also briefly presented.

(1) Evaluation of Corrosion Fatigue Degradation for Reactor Pressure Vessel and Primary Coolant Piping

The components constituting the primary coolant pressure boundary, such as RPV and primary coolant piping, are made of low alloy steel and stainless steel, which are subjected to stress cycles as they are used in the reactor coolant environment of high temperature and high pressure. Thus, it is required to predict the corrosion fatigue lives of these materials.

Corrosion fatigue data are being collected by various research institutions. However, sufficient data are not available at present, and in particular, there is little data on corrosion fatigue of aged materials.

From such point of view, it is intended in this test to collect material property data using aged (having thermal aging) materials and corrosion fatigue data of aged materials under high temperature/high pressure water environment.

The materials to be tested are two types of materials of low alloy steels, ASME SA 508 Cl.2 and SA 533 Gr.B Cl.1 (for BWR and PWR) concerning RPV, and twe types of materials of austenitic stainless steels, Type 304 (BWR) and Type 316 (PWR) concerning the primary coolant piping. The manufacture of materials has been completed in fiscal 1987. The material composition, manufacturing method, mechanical properties, etc., were simulated as close as possible to those of the early plants (1970s).

The test program consists of 1) the aging degradation simulation test in which the accelerated conditions corresponding 40 to 80 years of service are applied and 2) the corrosion environment test in which the corrosion fatigue data (fatigue crack initiation and propagation rate) of the thermally aged materials are to be obtained under BWR and PWR water environments. The fracture mechanics evaluation will be conducted in the final stage for RPV and primary coolant piping system as to corrosion fatigue.

(2) Evalauation of Thermal Aging Degradation of Duplex Stainless Steel for Primary Coolant and Pumps

The duplex cast stainless steels, consisting of austenitic phase and ferritic phase (10 to 25%) are used in the primary coolant piping (PWR), pumps and valves (BWR, PWR). It is known that the fracture toughness of these materials are reduced when subjected to high temperature for a long time due to precipitation of alpha'-phase (Cr rich), G-phase (Ni and Si rich) and carbides such as $M_{23}C_6$, resulting from decomposition of ferritic phase.

There are many unknown aspects concerning this phenomenon, such as the process of fracture toughness reduction at plant operating temperature, the effect of the ferrite content and manufacturing method on fracture toughness reduction, and verified prediction method has not been developed yet.

In view of these facts, the test program is mainly oriented to clarify the fracture toughness reduction phenomenon caused by the thermal aging of duplex stainless steels taking account of effects of ferrite content and production method, and to develop the prediction method on reduction of fracture toughness due to thermal aging.

The specimens served are piping materials (ASTM A351 CPF8M and CF8M), pump and valve materials (CF8) having different ferrite content and different manufacturing processes, and two welded metals (Type 316L) and HAZ. Thermal aging tempertures will range from 290℃ to 400℃, and aging will be conducted up to 24,000 hours at the maximum.

In the test program, fracture toughness reduction will be identified by fracture toughness test (JIC), Charpy impact test and the metallurgical examination.

(3) Evaluation of Irradiation SCC for BWR Reactor Internals

The objective of this test is to evaluate irradiation degradation of BWR reactor internals whose replacements are difficult. It is planned for this purpose to collect neutron irradiation embrittlement data, and to develop irradiation SCC growth model of Type 304 stainless steel for reactor internals evaluation.

The specimens are from actual reactors (control rod blade, SRM/IRM dry tube plunger, etc.) which have been irradiated up to nearly 10^{22} n/㎠(fast neutron level).

Also, the fracture toughness test will be conducted to collect embrittlement data, and in order to develop the irradiation SCC growth model, various tests, including material factor test, environmental factor test, mechanical factor test and model verification test will be conducted. The components to be evaluated will be the top guide, shroud, core plate, etc., which are deemed important for life evaluation.

(4) Evaluation of Irradiation Embrittlement on PWR Reactor Internals

The objective of this test is to evaluate irradiation degradation of PWR reactor internals whose replacements are difficult, and it is planned for this purpose to collect neutron irradiation embrittlement data and corrosion fatigue data of Type 304 and Type 316 stainless steels to develop life prediction method for reactor internals.

The specimens are also from actual reactors (flux thimble, fuel nozzle, thermal shield bolts, etc.) which have been irradiated up to nearly 10^{22}n/㎠ (fast neutron level).

The fracture toughness test and mechanical property test will be conducted to collect embrittlement data. And fatigue crack propagation test will be conducted in operating environment. The components to be evaluated will be the core baffle, core plate, upper core structures, core barrel, and thermal shield.

(5) Evaluation of Irradiation Embrittlement on Reactor Pressure Vessel

As mentioned before, JAPEIC is conducting the Pressurized Thermal Shock (PTS) Verification Test.

In this research, the predictive equation for fracture toughness reduction properties is being developed for extended service life based on the resultant data from accelerated irradiation fracture toughness testing (fast neutron level, nearly 10^{20}n/㎠), considering surveillance test data in both domestic and foreign countries.

In this project, the life of reactor pressure vessels will be evaluated in combination with the results of corrosion fatigue degradation and the achievement of the above research program, which is required for plant life extension.

4. Outline of Plant Life Extension Program

(1) Concept of Nuclear Power Plant Life Extension

Forty-year operating term lincenses are the maximum allowed for nuclear power plants, and may be renewed in the United States. In Japan, there is no legal stipulation defining the period of plant operation, and it is only stipulated in laws such as "Electric Utility Industry Law" that nuclear power plant must be subjected to periodical inspection with regular intervals. However, the components of nuclear power plant are designed based on 40 year design lives. Therefore, it is intended in developing technology for plant life extension to collect the aging degradation data to develop various life extension technologies by which the service lives can be extended beyond.

the initial design lives of components.

At present, there is not yet a definite concept concerning the methods of extending the life of nuclear power plant. However, a concept which is illustrated in Figure 1 can conceivable.

First, the life prediction for actual components and structures will be made based on operational data and condition monitoring data and future operation plan, using verified prediction models which will be prepared through aging degradation data base.

Next, when the plant life thereby determined is short of the target plant life, replacement/repair or measures for mitigating the effect of aging degradation will be enforced on the components/stuctures of the plant in question. In parallel, the economy of life extension must be evaluated, and for a unclear power plant in particular, the safety and reliability of longer service life must also be evaluated.

(2) Goals for The Program

The objectives of the technology development are to develop plant life prediction technology and to develop of life exetnsion technologies for critical components and systems. The specific goals for these development efforts are as follows:

〈1〉 Collection of component and material data which can be used in estimating the remaining lives of plants.

〈2〉 Proposal of methods for estimating plant remaining lives.

〈3〉 Proposal of technologies for replacement and repair for critical plant equipments and facilities.

〈4〉 Preparation of criteria (drafts) for reliability/safety in plant life extension.

〈5〉 Preparation of scenarios of plant life extension including economic considerations.

(3) Implementation Plan

The implementation schedule of the technology development program is illustrated in Figure 2. This program consists of three Phases, namely 〈1〉 the life extension feasibility study on components and concrete structures of nuclear power plants, 〈2〉 the verification tests and evaluation of life extension technologies, and 〈3〉 the overall evaluation.

The feasibility study of Phase I is already completed. In this phase, the survey of current status of technology developments concerning nuclear power plant life extension, selection of critical components which are important in plant life extension, and preliminary evaluation of plant life extension were conducted. The technology development items which are to be implemented in Phase II have also been identified in this phase.

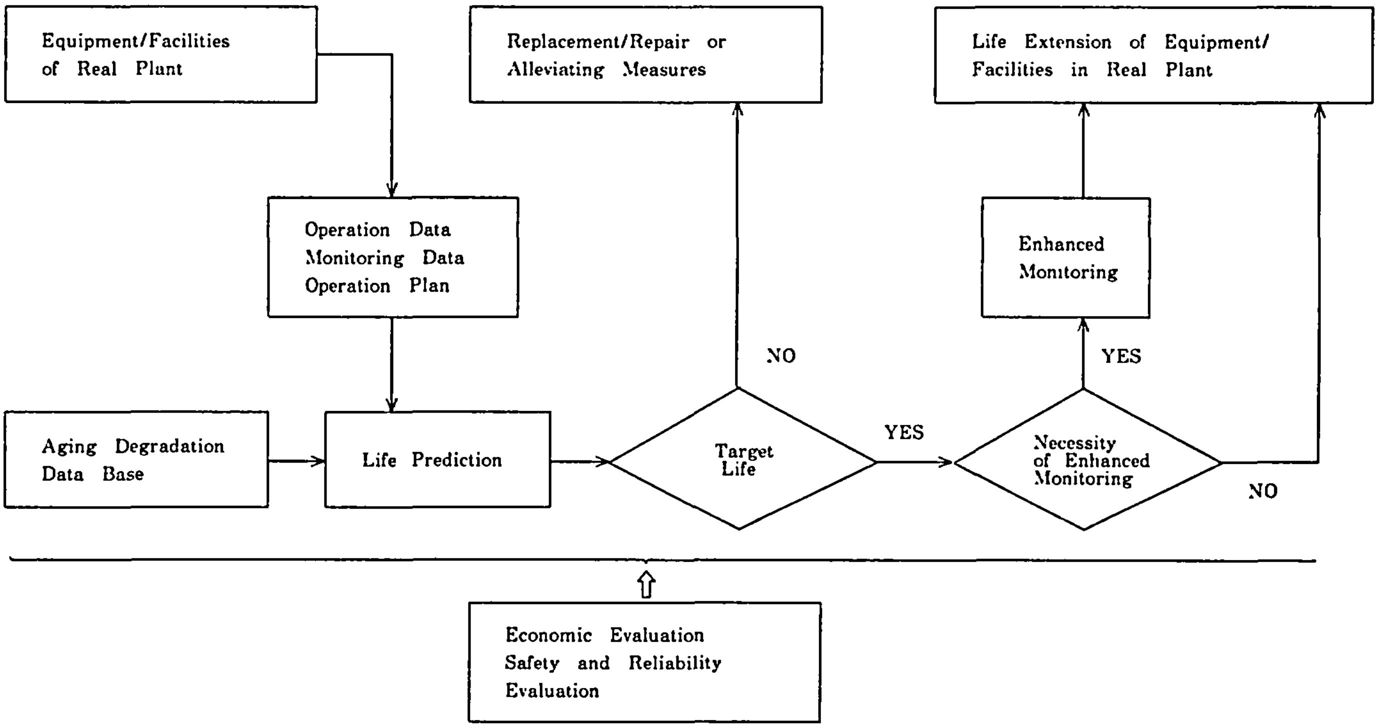


Figure 1 Concept of Nuclear Power Plant Life Extension

In Phase II, the verification tests, which have been selected in Phase I, and the evaluation of the life extension technologies will be implemented. In the verification tests, the aging degradation data. which are requited in life prediction of critical components, will be collected, and the material aging degradation tests required for development of life prediction methodology will be conducted, including, 〈1〉 the material test in corrosive environment on low alloy steel and stainless steel, 〈2〉thermal aging test of duplex stainless steel, 〈3〉fracture toughness test on irradiated stainless steel, 〈4〉 irradiation SCC test on stainless steel. In addition, it is planned to conduct 〈5〉 reactor pressure vessel surveillance specimen reconstitution verification test and 〈6〉 development of inspection/repair equipment for pressure vessel and reactor internals.

In fiscal 1987, the detailed research programs were formulated and some parts of verfication tests have been conducted.

In the life extension technology evaluation of Phase II, the aging degradation data will be collected for critical components/structures to conduct life predictions, and it is planned to develop system planning on aging degradation diagnosis and monitoring system and conceptual designs for replacement/repair of large components.

In the overall evaluation of Phase III, comprehensive evaluation studies will be conducted on the aging degradation data, life prediction methods, the life extension scenarios of nuclear power plants, and regulations and standards for nuclear power plant life extension.

No verification test is being planned on concrete structures, but similar studies as those for components/structures mentioned above will be conducted for technical evaluation of life extension.

5. Conclusion

We are executing MITI-sponsored aging research program for aiming at nuclear power plant life extension assessment. In the research program, corrosion fatigue test on low alloy steel and stainless steel, duplex stainless steel thermal aging test, and stainless steel irradiation SCC test and fracture toughness test were selected as research items which were considered to be important to plant life assessment based on the results of preliminary life assessment and their research program is presented.

In addition, the plant life extension program is also introduced.

We believe these research works will contribute to evaluating life extension as well as plant reliabilities.

6. Acknowledgment

The author is grateful to the Agency of Natural Resources and Energy, the Ministry of International Trade and Industry, and the Japan Power Engineering and Inspection Corporation for their supports and coopertion in preparing this manuscript.

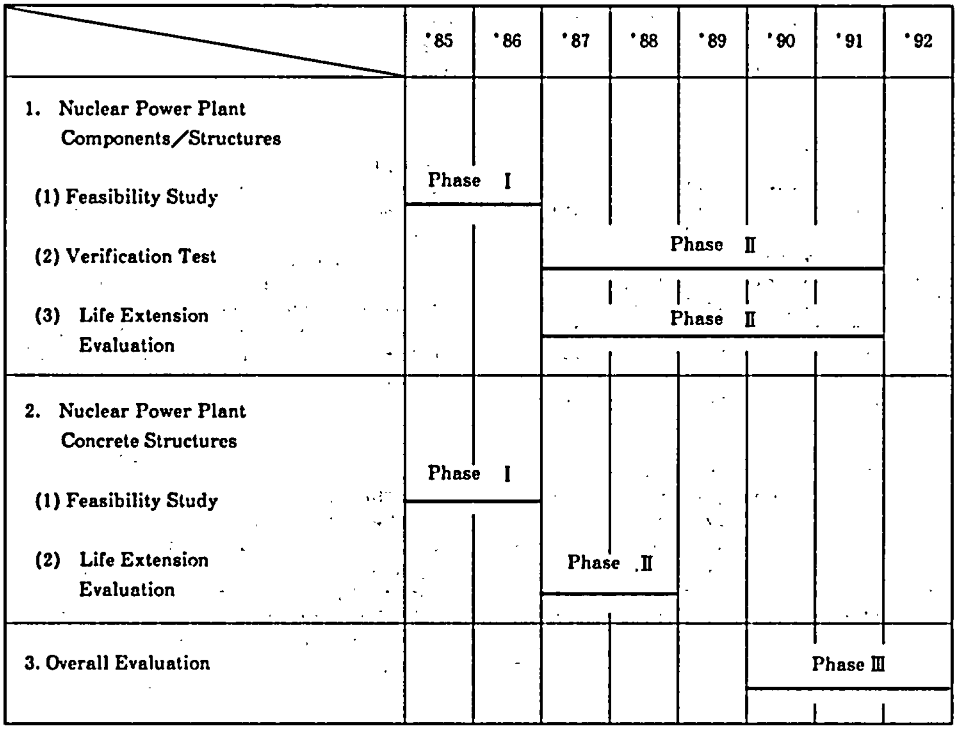


Figure 2 Nuclear Power Plant Life Extension Technology Development Implementation Schedule

A PROGRAMMATIC COMPARISON OF THE MONTICELLO PLANT LIFE EXTENSION AND NUCLEAR PLANT AGING RESEARCH PROGRAM

Gerry Neils
Terry Pickens
Daniel F. Lehnert

Abstract

This paper will draw a comparison between the Northern States Power (NSP) Company co-sponsored plant life extension (PLEX) program at the Monticello Nuclear Generating Plant and the NRC sponsored Nuclear Plant Aging Research (NPAR) Program. The comparative examination will identify key similarities and differences between the programs. Both of these programs focus on the effects and rate of aging; however, there are philosophical and programmatic differences which are resulting in differences of scope and priority. One of the differences which is addressed in this paper is the application of historical failure data as a basis for selecting the components, systems and structures to be studied and as a key measure of the effects of aging. The NPAR program essentially relies on the failure data to identify systems, and their respective system components, that are susceptible to age-related failures and warrant in-depth engineering studies. In the Monticello PLEX program, the failure data is only one of several physical, design, and operational factors which are considered in ensuring that resources are focused on the plant components, systems and structures required to preserve safety margins, to maintain availability goals and to achieve a cost beneficial extension of plant service life. The paper shows that the application of PLEX program "selection methodology" identifies only a subset of the NPAR selected components, systems and structures for in-depth engineering studies. This and other differences between the programs are important to recognize to ensure that their priorities, scope and results will effectively address industry and regulatory interests.

Introduction

The nuclear power generating industry is approaching a quarter century of commercial operating history. The current generation of nuclear power plants are well beyond the development stage and have reached a relative state of maturity. Today, nuclear generating facilities supply more than 16 percent of the U.S. power demand. In most cases, these plants are licensed to operate for a fixed initial operating term of 40 years. These facts and considerations have brought two issues to the forefront in the regulatory and utility industries.

The first issue is the assessment of age-related degradation and to resolve any consequential technical safety issues on affected plant safety components, systems and structures. This issue has been the focus of the NRC's Nuclear Plant Aging Research (NPAR) Program. The NPAR program defines a phased approach to research, identify and characterize the mechanisms of material and component age-related degradation during service and to apply the results in the regulatory process. In-depth studies and research have been proceeding for more than 3 years.

The second issue is to evaluate generation options when the current licensed term expires. In recent years, it has become increasingly difficult to license new plant sites, making full utilization of existing sites an important consideration for economic power generation in the future. Nuclear plant service life extension has, therefore, become a potentially viable alternative to new plant construction. Determining the viability of this option includes detailed engineering evaluations of how aging mechanisms are affecting the service lifetime limits for the plant. Northern States Power (NSP) Company and its Monticello Nuclear Generating Plant are the host utility and specific plant for the BWR Pilot Plant Life Extension (PLEX) Program. The Monticello PLEX program is providing the technology and data needed to maximize benefits from the plant, and to support decisions for extension of its operating license. Work on the program was initiated in 1983 and has proceeded through the feasibility and engineering stages and is entering the plant implementation phase.

Both of these programs are studying the effects and rate of aging. Seemingly, it would be expected that the scope of the NPAR program should be enveloped by the PLEX program because the PLEX program addresses a broader scope of issues. However, the programmatic comparison of the two programs presented in this paper shows that this is not entirely the case. There are philosophical and programmatic differences which result in differences of scope and priority. The differences between the programs are important to recognize to ensure that their priorities, scope and results will effectively address industry and regulatory interests, minimize duplication of effort, optimize available resources, and achieve coordinated goals.

The comparison of the NPAR and PLEX program is accomplished by first examining the concept of aging as it applies to both programs. Then the key programmatic aspects are presented and key differences noted. The aspects examined in this paper are the program objectives, the methods for selecting and prioritizing the elements of the plants, the evaluation approaches and the application of the results. The potential significance of the observed differences is offered in the conclusions of the paper.

Aging

The NPAR program has established a definition and the potential consequences of aging in NUREG-1144.[1] Aging is defined as the cumulative degradation that occurs with the passage of time in a component, system or structure. This degradation, if unchecked, is said to lead to a loss of function and an impairment of safety. The aging process is stated to begin as soon as a component or structure is produced and continues through its service life. NUREG-1144 provides that aging must be factored into the determination of safe operating lifetime limits, and that no nuclear plant should be considered immune from its effects.

The PLEX program did not need to specifically define the term aging. It is a fundamental premise of the PLEX program that aging will not unknowingly be allowed to be an accumulative effect. Aging is viewed as only one of several factors affecting the determination of prudent operating lifetime goals. Instead, the ongoing and continuous maintenance, inspection and renewal actions in the plant will be planned and formulated to control aging. The PLEX evaluations and implementation methods in the plant will provide the means to plan the necessary actions, monitor the results and thereby effectively manage the required residual service life margins. When the costs of these actions and the other factors affecting PLEX goals are predicted to exceed the costs of available alternatives for supplying generation, the limiting lifetime for safe operation of the plant will then have been determined.

Comparison of Program Objectives

The NPAR and PLEX programs share the common requirement to fully evaluate the effects and rate of aging. However, to draw a comparison between the programs, the objectives that determine how the aging evaluations will be applied are examined in this section of the paper.

The key implementation objectives of the NPAR program[1] emphasize the determination of the potential contribution to risk from failure of safety components, systems and structures as a result of aging. That is, the program focuses on ensuring that age-related degradation does not reduce the operational readiness of a plant's safety systems, components and structures, and that aging does not lead to failure of equipment in a manner that causes an accident or severe transient. The program's objectives also include the need to provide a systematic research effort and key information to enable the NRC to resolve technical safety issues and define its policy and regulatory position on plant life extension and license renewal.

The PLEX program[2] emphasizes the identification and investigation of the structures, systems and equipment which will have a primary influence on the life extension decision and goals for the plant. Implicit in the program's objectives are that the safety margins required to sustain the operational readiness of all the plant systems, including its safety systems, are preserved throughout the current and extended service life for the plant. Specifically, the plant life extension duration is to be based on the technical assessment of the components and structures identified as potentially life limiting. Technical and nontechnical issues are investigated to assure that any conditions or issues which could jeopardize the current licensed life of the plant and which will determine the ultimate plant service life limits are identified. Preventive actions, improvements and future work at the plant and industry levels are defined to manage the residual service life margins and to provide the necessary basis for license renewal. Establishing the cost-benefit of proceeding with plant life extension versus proceeding with an alternative generation approach is also a key objective of the program.

A comparative review of these NPAR and PLEX program objectives leads to several observations. As would be expected, the NPAR program focuses on only the plant safety equipment. The PLEX program does not make this distinction and addresses the entire plant in its scope. The PLEX program objectives do not single out the effects of aging. All technical issues and nontechnical issues, of which aging would certainly be a major consideration, are addressed in considering the PLEX option. Economic and technical considerations are also emphasized in the PLEX program. A clear understanding of the near-term and long-term effects of aging, and other technical plant and licensing issues, are needed to establish the cost-benefit of the PLEX option.

Methods to Select and Prioritize Components, Systems and Structures

The PLEX and NPAR programs recognized the need to select and prioritize the components, systems and structures for study. A nuclear plant is composed of more than 5,000 components. It would not be a prudent use of resources to equally study them all. Therefore, with the program objectives in mind, it was necessary for both programs to first define and apply a methodology to select, group and prioritize the components, systems and structures that should be studied in depth. It is interesting to compare the methods and results of the selection processes because this fundamental step essentially defines the scope and the philosophical point of view applied in the subsequent evaluations.

PLEX Program Approach

The PLEX program applied a comprehensive evaluation method to select and prioritize the components, systems and structures for study in the program. The evaluation method is represented in Figure 1. The approach results in ranking the plant elements with respect to their influence on plant service life goals, implementation decisions, and strategic planning considerations. The approach formulated a repeatable and defensible approach for component priority listing, and also provided a valuable basis for focusing the subsequent service life evaluations on the most influential issues. The key features of the PLEX approach are as described in the following paragraphs.

An initial identification and screening process was applied to logically group the elements of the plant and to subjectively rate them with respect to their importance to PLEX. Figure 2 helps to explain this step. The objective was to eliminate those elements having no impact on establishing service life limits (that is, routine maintenance items). The grouping process did not necessarily distinguish plant systems. It was recognized that component function, degradation, and aging were not necessarily system dependent. Therefore, the program scope could be considerably optimized by a logical grouping of the components and structures based on functional similarity, location in the plant and/or exposure to common environments.

Comprehensive selection and ranking criteria were applied by evaluation teams involving experienced personnel with a cross-section of industry expertise. The process considered thirty criteria in six categories involving feasibilty and impact of replacement/refurbishment actions, relative severity of the service conditions, importance to plant safety, service history, expected difficulty of achieving an extended service life and anticipated actions required to maintain reliability. The thirty criteria are displayed in Figure 3. The weighted criteria gave more influence to replacement feasibility, plant safety/regulatory and service condition considerations.

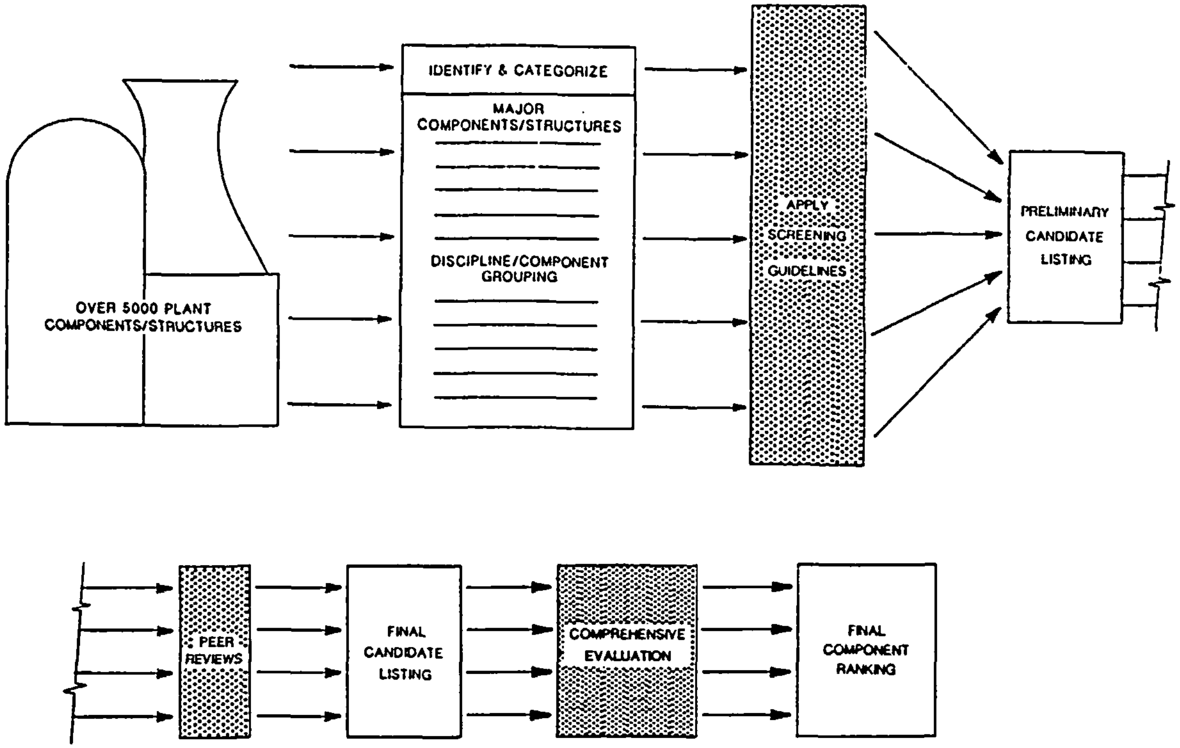
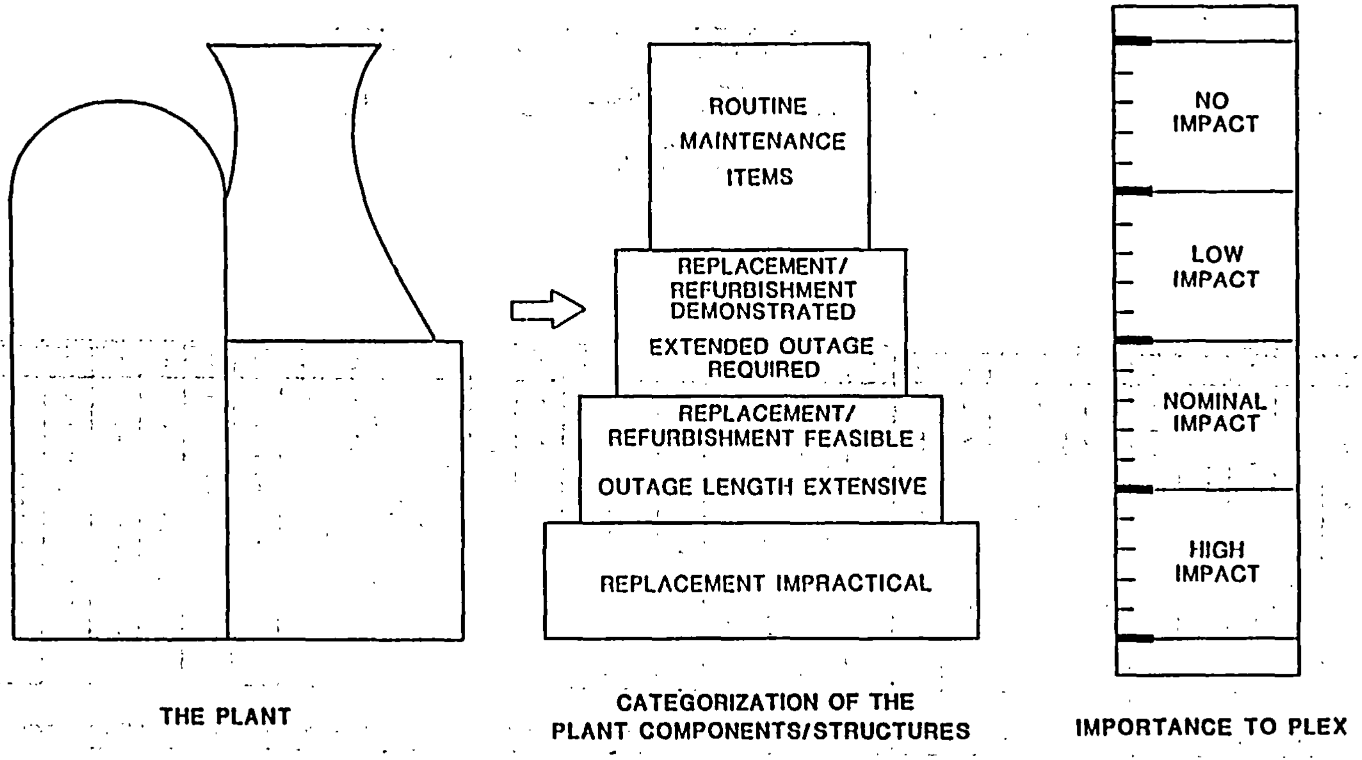


FIGURE 1

PLEX PROGRAM COMPONENT, SYSTEM AND STRUCTURES SELECTION AND PRIORITIZATION



THE PLANT

CATEGORIZATION OF THE PLANT COMPONENTS/STRUCTURES

IMPORTANCE TO PLEX

FIGURE 2
PLEX PROGRAM INITIAL IDENTIFICATION AND SCREENING PROCESS

CRITERIA LISTING & WEIGHTING FACTOR																													
GENERAL CRITERIA					SERVICE CONDITIONS						REGULATORY FACTORS					SERVICE HISTORY				POTENTIAL FOR SERVICE LIFE EXTENSION					RELIABILITY CONSIDERATIONS				
1. Replacement Feasibility	2. Outage Duration	3. Relative Replacement Cost	4. Adjacent Structures/Components	5. Disposal/Transportation	1. Brittle Fracture	2. Corrosion/Erosion	3. Dynamic Loading	4. Radiation	5. Thermal Shock/Creep	6. Synergistic Effects	1. Plant Licensing Basis	2. Safe Shutdown Path	3. Controversial Methods	4. New Codes and Regulations	5. Pending License Issues	1. Qualified High Maintenance	2. Potential Problem Indicated by Inspection	3. Already Replaced	4. Replacement is Planned	1. Qualified Vendors	2. Modification Technologies	3. In-situ Methods	4. Generic Test Data	5. Supporting Analytical Basis	1. Plant/System Derate	2. Enhanced Maintenance	3. Increased ISI/Surveillance	4. Add Redundancy or Spares	5. Preventive Measures
10	8	6	6	4	3	3	2	3	2	4	6	4	5	3	1	6	4	1	1	6	2	2	1	1	7	1	1	1	1

FIGURE 3

PLEX PROGRAM COMPREHENSIVE
EVALUATION SCREENING CRITERIA

NPAR Program Approach

The NPAR program essentially applied failure category information and PRA studies to select the safety components, systems and structures that would warrant in-depth studies. Expert opinion was also solicited through workshops to identify and obtain consensus of the components that impact safety and are most important, in terms of aging-related degradation. The approach results in the identification of the system and associated components most affected by aging parameters with respect to the impact on public health risk, core melt probability, and other risk measures. Some of the key aspects of the approach are discussed in the following paragraphs.

Information from the License Event Reports (LER's) and Nuclear Plant Reliability Data System (NPRDS) were used to categorize systems or components that are susceptible to aging-related failures.[3] Nine light water safety, support, and power conversion systems, including both PWR and BWR, were included in the information or failure survey. The nine systems are a subset of those finally included in the NPAR program.[1] The data of a further subset, the service water and Class 1E power distribution systems, were analyzed to establish a failure cause determination and to identify the aging mechanisms that caused component failures. Component aging reliability models were used to predict component failure rate data due to aging mechanisms.[4]

The results of PRA's were applied to evaluate the relationship between risk and component aging, and to measure the sensitivity of risk to changes in component failure rate.[5] The components are ranked with respect to their level of sensitivity for causing the greatest change in risk. Based on the three PRA's applied (Oconee, Calvert Cliffs and Grand Gulf), the most risk significant components were determined to be in the auxiliary feedwater system, reactor protection system and service water systems. Pumps, check valves, motor-operated valves, circuit breakers, and actuating circuits were the identified component types having the most potential risk impact.

Major light water reactor components of interest were identified and prioritized with respect to their role in preventing the release of fission products to the public.[5] This was a final input to the NPAR program selection process. The objective was to identify the major safety components that should be considered in life extension programs.

Comparison of Selection Approaches and Results

The methods applied by the NPAR and PLEX Program to select components, systems and structures for in-depth evaluation have some notable differences. The method applied in the NPAR program essentially focuses on the risk of aging and limits the scope to the safety elements of the plant. The PLEX program methodology sought to recognize the leading aspects of technical and economic feasibility to achieve an extended service life limit for the whole plant. In drawing a comparison of both approaches, the following observations are provided:

- The NPAR methodology was first designed to focus the program solely on the effects of aging, and was then expanded to also address the considerations of PLEX.
- The NPAR program strongly emphasizes the use of failure data, failure rate and unavailability predictions versus time and PRA evaluations to prioritize the elements of the plant. The PLEX program method includes operating history (failure data) and reliability (failure rate) as a subset of many other considerations in its ranking assessment. Replacement/refurbishment feasibility was a primary consideration in final ranking of the components and structures in the PLEX program.
- The PLEX program approach addressed the plant in terms of individual and groupings of components and structures. However, groups of components in the form of systems were not considered. It was viewed that recognizing, evaluating and preserving the key elements of a system would inherently lead to the preservation of associated system functions.
- The NPAR program stresses the need for future work in developing and proving risk models that consider the effects of aging through the plant's service life. It is reasoned that aging effects will change the risk ranking of the systems and components with time. Because of the multitude of factors or criteria considered in the PLEX program, the listing of "critical components," with respect to PLEX, would not be expected to change with time. The ultimate impact of aging would be replacement or refurbishment. The impact of these actions in technical and economical terms would remain relatively constant.

The comparative results of the two methodologies are shown in Table 1. The table lists the highest ranked components from both programs. For the NPAR program, these are the "Group 1" components and systems that apply to BWRs and selected major BWR plant elements.[1] The listed components and structures for the PLEX program are those that have been referred to as the "critical components" for PLEX. These are components, systems and structures that each program has focused its resources on to perform in-depth evaluations. For the NPAR program, there are Group 2 and 3 listings to be included in subsequent evaluations. The total PLEX program listing includes 120 components and structures. The remaining, or "noncritical," components are included in a lesser evaluation scope.

Table 1 shows that the PLEX listing is a subset of the NPAR program listing and compares more favorably with the NPAR major BWR plant elements subgroup. Most of NPAR component group items do not appear in the PLEX program critical component listing. These components ranked quite a bit lower[2] in the PLEX listing because they were viewed as easily replaceable items whose reliability could be effectively monitored by plant preventive maintenance and surveillance efforts.

Comparison of Evaluation Approaches

The evaluation of the selected components, systems and structures in the NPAR program is structured in two phases. The first phase[1] provides an interim aging assessment which includes a review of the hardware design, operating environment, and performance requirements; a survey of operating experience; and the current methods used for inspection, surveillance, monitoring, and maintenance and for qualifying end of life performance. The first phase is followed by in-depth, comprehensive aging assessments. This phase focuses on the validation of inspection, surveillance, and monitoring methods through laboratory and field testing activities and validating accelerated aging techniques. Its

TABLE 1 – COMPONENTS, SYSTEMS AND STRUCTURES SELECTED FOR IN-DEPTH EVALUATIONS

NPAR PROGRAM (BWR)	MONTICELLO PLEX PROGRAM
COMPONENTS ● MOTOR-OPERATED VALVES ● CHECK VALVES ● ELECTRIC MOTORS ● BATTERIES ● CHARGERS/INVERTERS ● SNUBBERS ● CIRCUIT BREAKERS AND RELAYS ● SOLENOID VALVES ● EMERGENCY DIESEL GENERATORS **SYSTEMS** ● HIGH PRESSURE EMERGENCY CORE COOLING SYSTEM ● LOW PRESSURE EMERGENCY CORE COOLING SYSTEM ● SERVICE WATER SYSTEM ● COMPONENT COOLING WATER SYSTEM ● REACTOR PROTECTION SYSTEM ● RESIDUAL HEAT REMOVAL SYSTEM/AUXILIARY HEAT REMOVAL SYSTEM ● CLASS 1E DISTRIBUTION SYSTEM **MAJOR BWR PLANT ELEMENTS** ● REACTOR PRESSURE VESSEL ● CONTAINMENT BASEMAT ● RECIRCULATION PIPING, SAFE ENDS, SAFETY SYSTEM PIPING ● RECIRCULATION PUMP BODY ● CONTROL ROD DRIVE MECHANISM ● CABLES AND CONNECTORS ● EMERGENCY DIESEL GENERATORS ● REACTOR PRESSURE VESSEL INTERNALS ● REACTOR PRESSURE VESSEL SUPPORT ● BIOLOGICAL SHIELD	● VESSEL PRESSURE BOUNDARY ● REACTOR PEDESTAL ● DRYWELL FOUNDATION ● BIOLOGICAL SHIELD ● REACTOR BUILDING BASEMAT ● FUEL POOL SLABS AND WALLS ● SACRIFICIAL SHIELD WALL ● DRYWELL METAL SHELL ● SUPPRESSION CHAMBER INCLUDING SUPPORTS ● RPV SUPPORT, GIRDER, AND BOLTS ● REACTOR BUILDING FLOOR SLABS AND WALLS ● PLANT CONTROL CENTER ● CRD HOUSINGS EXTERNAL ● CRD HOUSINGS INTERNAL ● TURBINE PEDESTAL ● SHROUD ● CORE SUPPORT PLATE ● CORE TOP AND BOTTOM GRID ● VENT LINES INCLUDING BELLOWS ● VENT HEADERS AND DOWNCOMERS ● EMERGENCY DIESEL GENERATORS ● REACTOR RECIRCULATION LINES ● JET PUMPS ● NOZZLE SAFE ENDS ● ECCS PIPING INSIDE CONTAINMENT ● MAIN STEAM LINES INSIDE CONTAINMENT ● FEEDWATER PIPING INSIDE CONTAINMENT

objectives also include the development of models to simulate degradation, in-situ aging assessment and testing of naturally aged equipment. The two-phased approach is applied to all components and systems in the NPAR program.

The evaluation of the "critical components" in the PLEX program assessed the historical and current condition of the component, quantified the component's residual service life, and identified effective follow-on actions to improve or assure the predicted service life. The key elements of the evaluation are displayed in Figure 4. The evaluations provide a thorough assessment of potentially life limiting causes and effects. Physical life predictions were made on the basis of existing plant and industry data, simplified failure models, in-plant testing and inspections, and engineering judgement. The safety margins historically required by the code and for postulated emergency or accident scenarios were preserved in the residual service life assessments and in establishing component service life limits. Therefore, extended plant operation does not mean safety margins are reduced or that risks are increased. The results of the critical component evaluations, as a group, establish the service life goals for the plant. The follow-on actions are comprised of technical evaluations; plant inspection, testing and planned renewal actions; and research and development activities. These actions are all designed to support the conclusions of the component studies, and to support the technical, economic and licensing basis for extending licensed service life of the plant. The detailed evaluations of the plant's critical components for PLEX provide a thorough examination of the plant's degradation mechanisms. These results are also applied to more efficiently evaluate the remainder of the plant's components.

The general approach of the NPAR and PLEX program component, system and structural investigations are very similar. The PLEX program utilized a single-phased approach to essentially accomplish the elements of the two-phased NPAR assessments. Some of the NPAR Phase 2 activities, such as testing of naturally aged components, would be singularly identified and scoped as follow-on actions in the PLEX program.

Application of Program Results

The NPAR program results are intended to provide the basis for regulatory decisions regarding the continued safe operation of nuclear plants of all ages.[1] The NRC anticipates use of the results during the review of license renewal requests. The information is also expected to benefit plant maintenance practices by making them more effective. Some of the more specific applications of the results that apply to operating plants are to:

- Support NRR/RES in resolving generic safety issues involving aged safety systems, support systems, and electrical and mechanical components.
- Evaluate and recommend surveillance and maintenance methods needed to monitor age-related degradation and to support license renewal.
- Develop technical data and provide recommendations useful to AEOD and NRR, during plant inspections and review of license renewal requests, for developing plant performance indicators.
- Provide information for developing in-service inspection procedures suitable for aged components, systems and structures.

Phase 1 and some Phase 2 results have been reported for several of the component category evaluations. Some of the more advanced efforts included the studies of motor-operated valves, check valves, chargers/inverters, power-operated relief valves (PORV), snubbers and diesel generators. Activities leading to application of available results in the regulatory process have essentially just begun. Most of the effort in this regard will not be completed until 1989 or 1990. Application efforts for the PORV and snubbers have been completed. For the snubbers, the application of the study results has led to information exchange with code committees and a value impact analysis reflecting the reduction of snubbers (Draft Regulatory Guide SC708-4).

The results of the Monticello PLEX program provide short-term and long-term input to issues and programmatic needs of PLEX technical feasibility, economic justification, relicensing basis and plant implementation. In the short term, the results have demonstrated that there are no technical obstacles to PLEX and the benefit to cost ratio is very favorable for PLEX when compared to other generation options. Substantial increases of plant service life have been demonstrated to be technically achievable. Actions and resources can be effectively identified and planned to maintain both safety and economic goals. The plant cycle management chart (Figure 5) is the tool for maintaining this perspective. For the long-term, the results have established the means to acquire the necessary information and data to monitor the effectiveness of the plant's efforts to manage the residual service life of the plant and to provide the input for a relicensing submittal. Practical improvements to maintenance, surveillance and inspection efforts are required to enhance service life evaluations and to maintain margin awareness. The improvements will allow the plant to track and monitor the key parameters against practical threshold criteria (established in the service life evaluations) to allow advanced and timely preventive actions. The favorable economics of PLEX would justify scheduled replacements/refurbishments of many plant commodity items. That is, when their performance becomes suspect, they can be replaced or other prudent corrective actions elected. Ongoing and detailed studies of risk versus age need not be a required course of action. Advanced knowledge of the activities and actions that will have to be implemented by the plant to sustain safety margins and service life goals is a key benefit of the PLEX results. These activities and actions will negate the effects of age-related degradation. Improved margin awareness and management is expected to lead to increases in plant availability, and improvements in issue identification and resolution.

Conclusion

This paper has compared the programmatic features of the NPAR and the Monticello PLEX Programs. The comparison has demonstrated that both programs share the goal to evaluate effects and rate of aging. There are also close similarities in the scope and methods of the evaluations of the components selected for in-depth study. However, there are programmatic and philosophical differences that affect the priority in which components are selected for study, the evaluation results and the intended application of the results. Some of the more notable differences recognized in this paper are summarized below:

FIGURE 4
KEY ELEMENTS OF A PLEX PROGRAM CRITICAL COMPONENT EVALUATION

HISTORY & LIMITATIONS

- CODES, STANDARDS AND REGULATIONS
- EQUIPMENT QUALIFICATION
- DESIGN BASIS AND RELEVANT TEST DATA
- OPERATING HISTORY
- MAINTENANCE HISTORY

RESIDUAL LIFE

- DEBILITATING EFFECTS AND EVENTS (I.E., RADIATION, FATIGUE, CHEMICAL ATTACK, SYNERGISTIC EFFECTS, WEAR, ETC.)
- MONITORING AND DATA REQUIREMENTS (ISI, SURVEILLANCE, MAINTENANCE AND RECORDS)
- LIFE PREDICTION

IMPLEMENTATION FACTORS

- POTENTIAL REVISIONS TO OPERATING AND MAINTENANCE PROCEDURES
- POTENTIAL MODIFICATIONS
- COST/BENEFIT ANALYSIS
- SPECIAL CONSIDERATIONS

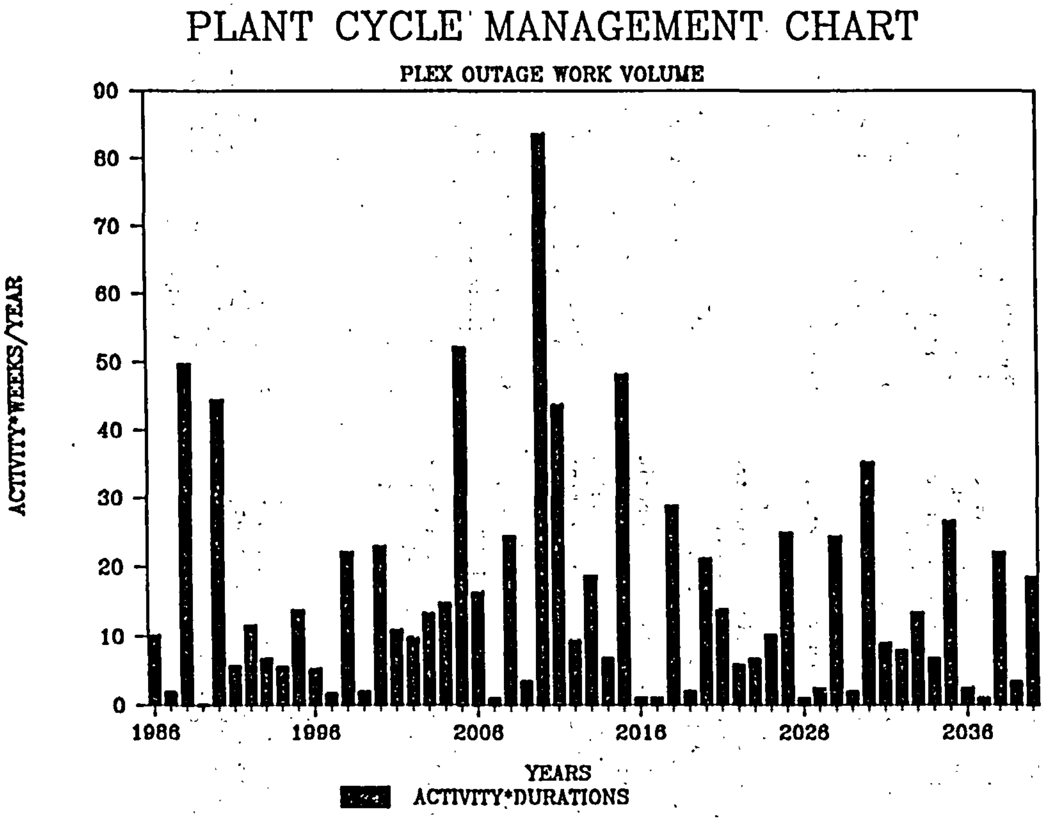

PLANT CYCLE MANAGEMENT CHART
PLEX OUTAGE WORK VOLUME
ACTIVITY*WEEKS/YEAR
90
80
70
60
50
40
30
20
10
0
1986
1996
2006
2016
2026
2036
YEARS
ACTIVITY*DURATIONS

FIGURE 5

- The NPAR program studies and research focus solely on the effects and rate of aging. In the PLEX program, aging is one of several factors affecting the safe and economic lifetime limit for the plant.

- The NPAR program defines aging as cumulative effect that will lead to a loss of function and safety implications. The PLEX program embraces the premise that aging can be prudently managed to avoid any cumulative effects, and thereby preserve required service life and safety margins.

- To demonstrate the viability of a PLEX program, the entire plant is evaluated. The NPAR program focuses on just the safety-related components, systems and structures of the plant.

- The NPAR philosophy looks at failure history which is overwhelmed by "childhood diseases." Most of these generic problems have been addressed and are weeded out as time goes on. PLEX looks ahead to identify "tomorrow's problems" by projecting observed degradation and rate of aging.

- The NPAR program strongly relies on risk ranking, based on failure and failure rate data, to prioritize and select elements of the plant for in-depth study. This approach anticipates that the risk ranking will change with time due to the effects of aging. The PLEX program applies a comprehensive screening criteria that includes failure data, and many other technical and economic considerations, in selecting the key elements of the plant for in-depth study. The approach results in an importance ranking that would remain constant with time.

- Many of the NPAR components selected for in-depth study do not appear in the corresponding PLEX program priority listing of components to be studied in depth. The PLEX program assigns a lower priority to these components because renewal and/or preventive actions are more easily achievable and are prudent alternatives.

- The intended application of the NPAR results strongly emphasizes the need for applying failure data in ongoing and detailed studies of risk versus age. The PLEX program results demonstrate that original design margins for the key elements of the plant have, in most cases, provided ample opportunity to extend service life limits of the plant without encroaching on safety margins required by code or regulation. This, combined with the favorable economics of PLEX, would allow renewal or other preventive measures to be a viable course of action for addressing the bulk of the plant's components. Time-dependent PRAs and risk models to account for the effects of aging throughout the plant's service life need not be a requirement.

These differences are quite significant with respect to interests of the industry and the regulator. By its nature, the breadth and scope of the PLEX program should envelope the NPAR program. This paper has shown that this is not entirely the case. The NPAR program is applying resources to in-depth studies that will not be a factor in determining safe operating limits for a plant implementing a PLEX program. The prescriptive application of the NPAR results in time-dependent risk models will have practical value to the industry. It is believed that the NPAR program can benefit from the PLEX Program accomplishments to resolve the differences discussed in this paper and thereby better serve all interests.

References

1. Nuclear Plant Aging Research (NPAR) Program Plan, U.S. Nuclear Regulatory Commission, September 1987, NUREG-1144, Revision 1.

2. BWR Pilot Plant Life Extension Study at the Monticello Plant: Phase 1, U.S. Department of Energy and Electric Power Research Institute, May 1987, NP-5181M.

3. An Aging Failure Survey of Light Water Reactor Safety Systems and Components, Idaho National Engineering Laboratory, July 1987, NUREG/CR-4747, Volume 1.

4. Risk Evaluations of Aging Phenomena: The Linear Aging Reliability Model and Its Extensions, Idaho National Engineering Laboratory, April 1987, NUREG/CR-4769.

5. Importance Ranking Based on Aging Considerations of Components Included in Probabilistic Risk Assessments, Battelle's Columbus Laboratories, Pacific Northwest Laboratory, April 1985, NUREG/CR-4144.

DISCUSSION OF
"A Programmatic Comparison of the Monticello Plant Life Extension and Nuclear Plant Aging Research Program," by G. Neils, T. Pickens, D. Lehnert

by
M. Vagins, J.P. Vora

The authors' interest in the NRC-sponsored Nuclear Plant Aging Research (NPAR) program is very much appreciated. In their paper the authors have attempted to provide a programmatic comparison of the Monticello Plant Life Extension and the NPAR program. The paper portrays a very narrow view of the NPAR program and attempts to compare apples with oranges. It restricts the NPAR review to a small fraction of the overall scope of NPAR work. Additionally, the authors have attempted to compare a plant-specific life extension program, which is primarily dictated by economic considerations, to the NRC NPAR program which addresses technical safety issues for <u>all</u> operating nuclear plants of <u>all</u> ages. Furthermore, the authors have failed to recognize the most significant tasks of the NPAR program and omitted the discussion on potential application of NPAR results in the regulatory process. The following discussion should be useful in clarifying some of the misunderstandings and misinterpretations the authors have had about the NPAR program.

- In addition to providing the technical basis for license renewal, NPAR addresses technical safety issues related to aging for the <u>current operating license period of 40 years</u> for all operating nuclear power plants.

- It is our understanding that aging-induced component, system, and structural degradations and failures are not addressed currently in the Monticello PLEX Program. It is appropriate that the NRC be concerned about the potential increase in risk associated with aging during the initial 40-year license period as well as during any license renewal period.

- For many safety system components, no sound technical basis for test intervals, surveillance, maintenance, refurbishment, and replacement policies exist. The effects of aging per se are not addressed. The NPAR program addresses these issues, PLEX programs do not.

- For most safety system components proper simulation of aging processes is missing during qualifications. NPAR addresses this issue, PLEX programs do not.

- The NPAR projects are studying many components and systems for which renewal and/or preventive actions are achievable and which are not being studied in the PLEX program. These components reside in the important plant safety systems; strongly influence plant risk; and may, at some plants, not be properly maintained or replaced. A good understanding of their failure modes and effects and improved IS&M standards can significantly reduce the overall plant risk.

- NPAR addresses and provides technical bases for the resolution of Generic Safety Issues such as II.E.6.1, "In Situ Testing of Valves," B.56, "Diesel Reliability," and GSI 70, "PORV/BV Reliability."

- Special topics which are being addressed in the NPAR program also include:
  - Risk Evaluation of Significant Aging Effects and Component Prioritization
  - Aging Evaluation at Shippingport
  - Residual Lifetime Evaluation of Major LWR Components and Structures
  - Technical Bases for Development of:
    - Regulatory Guides
    - Standard Review Plan
    - Criteria for License Renewal
    - Maintenance Rulemaking
  - Reviews of Technical Specifications From Aging Perspective
  - Reviews of Data Needs and Recordkeeping From Aging Perspective.

Despite the apparent programmmatic and philosophical differences in the NRC NPAR and Monticello PLEX programs, we conclude that these programs are very complementary and will both help us reach our objectives of providing safe, reliable nuclear power.

UNDERSTANDING AND MANAGING AGING AND MAINTENANCE

J. P. Vora, J. J. Burns

ABSTRACT

Aging of materials in the presence of operating stressors and environment is normal and universal in nature. Aging occurs in all large engineered complexes, including aerospace, petrochemical, and fossil fuel power plants, as well as nuclear power plants (NPPs). In NPPs, which are designed, operated, and maintained with differing philosophies and practices, aging affects all reactor structures, systems, and components. If these effects are not managed correctly, aging can increase the risks to public health and safety. Therefore, the aging of NPPs must be rigorously managed.

This paper describes a systematic approach, encompassing the technical elements, to managing aging in NPPs. In this context, the paper addresses issues related to what to maintain, when to maintain, and how to maintain plant structures, systems, and components.

INTRODUCTION

Aging, if it is not managed properly, affects all reactor structures, systems, and components' operational safety, and it has the potential to increase risks to public health and safety. There are significant uncertainties with regard to aging-related degradation processes which affect key components and structures and about the way such degradation can be detected and managed before safety is impaired. Aging is a key concern in the operation of plants and will clearly be a crucial issue in any assessment of the safety implications of license renewal. Specifically, there is concern that simultaneous multiple failures of age-related components could occur during a transient or accident compromising safety system function.

The operating experience indicates that component failures have occurred because of corrosion, radiation, thermally induced embrittlement of electrical insulation, pitting of electrical contacts, surface erosion, metal fatigue, oxidation, creep, binding, and wear. Therefore, as operating reactors advance in age towards the normal design life of 40 years and as we contemplate extended life the major regulatory safety issues that exist are: (i) Will aging of plant systems and components result in common mode failures that will weaken the defense-in-depth strategy, lead to an accident, or render the redundant but aged safety equipment, needed in accident mitigation, to be inoperable. (ii) How to assure operational readiness of aged equipment through the operating years.

UNDERSTANDING PLANT AGING

It is essential to understand the aging processes that occur in a system or a component before the age-related degradation can be effectively managed. To understand these aging processes, one must review the system or component design, fabrication, installation testing, inservice operation, and maintenance cycles. This is due to the fact that each of these elements in the life cycle of a system or component involves the interaction between its operational environment and the stressors associated with its materials. Some specific elements that impact on the plant aging are presented in Table I.

Materials

All materials used in the fabrication of components that make up the systems of a nuclear power plant experience age-related degradation of some sort. Of concern is, can this time dependent degradation affect the operating function of the safety-related components and systems such that safe operation of the nuclear power plant is reduced below acceptable limits.

It is important to understand how and at what rate the metallic, non-metallic, and composites of materials used in safety-related components degrade with time, and how this degradation can be mitigated or managed to assure the operational readiness of the component. This knowledge of material behavior response is extremely important not only in the design phase and the application of the component and plant systems, but is also important in the development of the plant quality assurance, plant inspections, condition monitoring, and maintenance programs. The more we know about the age-related behavior of materials and the use of this knowledge in the design and operation of components using these materials, the more confidence we will have in assuring the components' lifetime behavior and plant operational safety.

Environments

It is important to understand how various environments affect the age degradation of safety systems and components. Here environments include the operating environments (air density, humidity, and temperature) within the plant or within a storage facility, chemicals induced onto the material (pollutants, acids, and fluids), radiation, and temperature. Environmental effects by themselves can cause aging such as material corrosion or combine with other degrading elements (e.g., material choice, environment, heat, and loading stress to cause intergranular stress corrosion cracking). Understanding also is needed as to how the severity of the environments can affect the rate of age degradation.

Stressors

Of all the elements that can affect the age degradation of nuclear power plant components and systems, the associated stressors are generally the most difficult to understand. Stressors take on many forms (e.g., mechanical, electrical, thermal), and some of them have their origins in the material and component fabrication, assembly, transportation, installation, operation, testing, and maintenance. A component designer, fabricator, operator, and maintainer must appreciate and understand how stressors can degrade the operational capabilities of a component and system.

Mechanical stressors are generally associated with the physical movements and dimensional changes. The operation of a component either from its normal operational modes or accident conditions will produce time dependent mechanical stresses. These stresses are induced by dynamic and acceleration loads, developing internal or external pressures, impact, vibration loads, temperature changes, component test loads, and seismically induced motions.

The operational motion of active components (e.g., valve operation and pump rotations) will produce time dependent distortions and inertia stresses. These motions will also produce component wear. The effects of these distortions and wear on the age degradation of a component are generally understood, but the rate of degradation is usually estimated only from the analysis of inservice monitoring data, inspection, and maintenance information. It is observed that the proper maintenance of a component can negate much or all of the degradation incurred within a component. Internal and external pressure loads approaching design and accident conditions also can produce very high stresses which can produce distortions and with a sufficient number of applications can produce strain hardening and initiate fatigue damage to component materials. If these stresses are combined with vibration and thermal stresses, measurable degradation can incur in a period of time which can be small when compared to the operational life of the component. Both the component designer and its operator must understand the significance of pressure loading on a component in degrading its operational margins.

Impact, due to a seismic event, on a component at any point during its operational life can induce immediate damage to the component. Even though the component may not fail during the impact, its design and functional capability may be degraded such that its operational life may be shortened. The extent of the potential damage in a component resulting from external impact must be understood to adequately predict component age degradation. For example, in spite of some dimensional changes in winding spacers of a transformer during a seismic event, a transformer may continue to function but may fail in later years of its operating life.

Vibrational loads, if large enough, can appreciably add to the fatigue damage in a component. Methods of analyzing the vibrational fatigue damage incurred in a component are currently available. However, the results of such analysis can contain many uncertainties. These uncertainties are associated with the component material fatigue properties and the distribution and cyclic level of the induced dynamic stresses. The vibrational stresses also are induced from plant operational modes, and during transportation, vibrations can introduce very high cyclic loads if the component is not properly isolated from ground or seismic vibrations. Care must be taken to understand the source of vibrational loads that may be developed during the operational life of the component, the distribution of the stresses incurred, and the endurance limit of the materials in the component. A practical example of stress enhancement is during the fast start of emergency diesel generators.

During the operation of a diesel generator system, thermal gradients are developed. These gradients cause relative expansion of the contained materials which can affect tolerances and possible dimensional interferences. It has been concluded, however, that much of the wear and aging effects in a diesel generator occur during the fast startup of the system when the gradients are substantial. Also, the shorter the time between starting to full engine power, the greater will be the susceptibility to damage.

The nature, speed, and frequency of tests will impact the reliability of the component or system.

Electrical stressors are induced across the insulating materials used in the fabrication of electrical and electromechanical devices. Both the passive (cables, connectors, electrical penetrations, transformers, terminal boards, etc.) and the active (motors, circuit-breakers, relays, voltage and current activated devices, etc.) electrical components experience voltage gradients, depending upon their applications, during normal operation and testing. Of concern are the higher levels of electrical stressors which are generated during switching operations and during the accident and post-accident situations especially, when the dielectric voltage withstand capability of the insulating systems may degrade with age.

The nature of the electrical voltage stressors may vary depending upon the design and the functional application of the device. The nature of the voltage waveform could be d.c., 50-60 Hz a.c., fast micro-second transient, or slow switching transient. More importantly, the severe voltage gradients are experienced when a device is subjected to various combinations of these voltages superimposed at the same time.

It is, therefore, important to accurately assess the magnitude and duration (signatures) of voltage and current-related stressors in plant electrical components and systems during (i) normal operating conditions, (ii) test conditions, and (iii) accident and post-accident situations.

Thermal stressors induced into a component or an assembly of components can be attributed to the fact that the mechanical and physical properties of the materials vary with temperature. Different materials expand at different rates when heated and their stiffness varies with temperature. These expansions may be resisted internally or by an interference by adjacent component surfaces. This resistance and the induced temperature change in material stiffnesses and strengths will result in time dependent component thermal stressors. The thermal stressors when combined with the effects of other stressors can result in a time dependent aging degradation of the different components. Typical of such degradation are the thermal fatigue cracks that have developed in plant high-temperature coolant water piping and nozzles.

Materials, environment, and stressors interaction. The age-related degradation of the components, systems, and structures is a time dependent phenomenon and depends upon the interactions of materials, environment, and stressors. While performing aging assessments, one cannot and must not ignore the influence and effects of these interactions. Every component and system under study must be evaluated in this context. For example, in evaluating an aged insulating material and its voltage withstand capability, not only must we consider the effects of temperature and radiation, but we must understand the influence on aging of humidity, oxidation, contamination, plus the simultaneous effects of electrical and mechanical stressors acting under normal operating, testing, accident, and post-accident situations. For a mechanical device or a fluid-mechanical system the material-environment-stressor interactions result in corrosion, errosion, fatigue crack growth, creep, and mechanical wear.

Once the interactions that cause components and systems degradation are understood then they must be managed to assure that aged components and systems will adequately perform their design safety functions. Based upon these interactions, measures can be developed to monitor the key safety-related components and systems to detect the degree of degradation and manage aging through proper inspection, surveillance, condition monitoring, maintenance, and replacement programs.

It is recognized that some of the aforementioned assessments and analyses are difficult. Nevertheless, efforts must be made to acquire and record as much of this knowledge on risk significant and prioritized sets of components and systems as possible. It is only then we can develop confidence toward aging assessment and residual lifetime evaluation and determine the operational readiness of the aged components and systems.

MANAGEMENT OF AGE DEGRADATION

Risk Significance and Component Prioritization

Time dependent calculations that take into account the effects of aging are necessary to identify and prioritize risk significant components, systems, and structures, then in turn to develop programs to understand and manage aging in those prioritized components, systems, and structures. Techniques for performing time dependent risk or core-melt probability calculations need to be developed. The task to manage, including the allocation of proper resources, the effects of age degradation in nuclear power plants becomes much easier if the risk significant components and systems that will degrade during a normal operational or extended life are identified and prioritized. To accomplish this, a program that includes a hybrid approach (that is, a deterministic approach in conjunction with a probabilistic approach) must be developed. We must utilize the knowledge gained from engineering designs, applications, tests, and operating experience. Also, data from in situ assessments, condition monitoring, recordkeeping, and post-service examination and tests are essential for developing suitable deterministic models and for risk assessments and component prioritization. Expert panel workshops also are recommended in the prioritization process. Of particular concern are those components and systems, not easily or routinely inspected and maintained, that may degrade with age and impact upon plant safety.

Maintenance to Manage Aging

Condition monitoring, trending, recordkeeping, and maintenance programs at nuclear power plants are extremely important to manage aging and reliability assurance. Effective maintenance programs will require understanding of what to maintain, when to maintain, and how to maintain plant systems, components, and structures. The earlier discussions on understanding aging provide clues on what to maintain.

The key steps in determining when to maintain and how to maintain a specific system, a structure, or a component from an aging perspective are:

- identify performance measures or functional indicators for each of the risk significant and prioritized specific systems, structures, or components which would give an indication of its health at the time of observation
- then, identify methods to detect performance measures or functional indicators in incipient state prior to failures
- trend performance measures or functional indicators for each specific system, structure, or component under observation and analyze the impact of rate of change
- develop a library of data, information, guidelines, and criteria
- determine minimum functional capability at the end of normal service life
- determine minimum functional final capability to mitigate an accident
- interpret, analyze, make decision for maintenance or replacement.

Maintenance programs are needed, both predictive and preventive, to manage aging. A program to manage aging will aid in making decisions for corrective maintenance, quality assurance and quality control, engineering support, and plant modifications.

NRC Aging Research Program

The NRC nuclear plant aging research (NPAR) program is directed toward gaining knowledge and understanding of degradation processes within nuclear power plants. This hardware-oriented engineering program is a rigorous and systematic investigation into the potentially adverse effects of aging on plant components, systems, and structures during the period of normal licensed plant operation, as well as the period of extended plant life, that may be requested in utility applications for license renewals.

Emphasis has been placed on identifying and characterizing the mechanisms of material and component degradation during service and utilizing research results in the regulatory process. The research includes evaluating methods of inspection, surveillance, condition monitoring, and maintenance as a means of managing aging effects that may impact safe plant operation. Specifically, the goals of the program are:

- Identify and characterize aging effects that, if unchecked, could cause degradation of components, systems, and structures and thereby impair plant safety.
- Identify methods of inspection, surveillance, and monitoring, and evaluate residual life of components, systems, and structures that will ensure timely detection of significant aging effects before loss of safety function.
- Evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.

The NPAR program is based on a phased approach to research. The objectives of the Phase I studies are: to identify and characterize aging and wear effects; to identify failure modes and causes attributable to aging; and to identify measurable performance parameters, including functional indicators. The functional indicators have a potential use in assessing operational readiness of a component, structure, or system in establishing degradation trends, and in detecting incipient failures. The objectives of the Phase II studies are: perform indepth engineering studies and aging assessments based on in situ measurements; perform post-service examinations and tests of naturally aged/degraded components; identify improved methods for inspection, surveillance, and monitoring, or for evaluating residual life; and make recommendations for utilizing research results in the regulatory process.

Risk Significance of Aging. Aging models and risk assessment methodologies require development to provide quantitative determination of the effect that aging has on safety. The major activities within this research element are: (i) aging model development, including the treatments of active components, passive structures, and the influence and effects of testing and maintenance; (ii) failure data analysis; (iii) engineering information analysis; (iv) uncertainty analysis; (v) application and demonstration of the risk assessment methodology; and (vi) the development of procedures and guidelines for treatment of aging in probabilistic risk assessments (PRAs).

Aging/Systems Interaction Study. Aging system interaction study is essential to determine how aging is affecting component and system unavailability and to establish the relative contribution to risk from age-related component and system failures. This element of the aging research program will facilitate the prioritization of plant safety systems and components for indepth engineering studies. Then, generate guidelines and recommendations for inspection and maintenance to alleviate aging concerns.

The Integrity of Primary System Components Program. This element of the aging research is focused on aging issues for the materials and components of LWR primary systems. The goal of the program is to provide the confirmatory technical basis for regulatory decisions on the safe operation of reactor vessels, primary system piping, steam generators, and improvements in the techniques and equipment required for nondestructive in-service inspection of these components.

Electrical and Mechanical Components. Some 30 categories of components (e.g., pumps, motors, valves, cables) are the subjects of current and planned research being studied by five national laboratories and several private institutions and organizations.

The results (a synopsis) of the NPAR approach to understanding and managing aging and maintenance are illustrated in Figures 1 through 6 for some specific risk significant components and structures. They are PWR Reactor Pressure Vessel (Figure 1), PWR RCS Piping and Nozzles (Figure 2), BWR Mark I Containment (Figure 3), Diesel Generators (Figure 4), Motor-Operated Valves (Figure 5), and Cable System (Figure 6). These illustrations are not intended to include all relevant information. They are simply included here to illustrate an NPAR approach useful to address any component, system, or structure of interest, to identify the age-related degradation sites within the component boundary of interest, and generate recommendations for maintenance and mitigation of aging. More comprehensive aging-related information on each component is provided in the NRC NUREG reports.

SUMMARY AND CONCLUSION

Considerable importance must be placed on understanding and managing aging effects within operating nuclear power plants. The primary safety concerns are the potential reduction of defense in depth and common mode failures attributable to aging and aging-system interactions.

To understand the significance of plant aging on safety is to understand how plant risk changes because of the aging effects. It must be realized that this risk is time dependent. When the properties of structures and components degrade, then their reliabilities can degrade. The reliability is determined by such quantities as the frequency of an initiating event (such as a pipe break), the failure rate, and the unavailability. When the reliabilities of structures and components degrade, the safety and risk of the plant can be adversely affected.

The relationship whereby mechanisms cause property *degradations*, which cause *reliability degradations*, which in turn affect plant safety, is the connection between aging and plant safety. To understand and control the effects of aging on plant safety, it is important to model the individual relationships involved in the overall connection between aging and safety.

To understand aging effects on components and systems is to understand the effects of the environment, coupled with the internally generated and applied external stressors on the component materials within its operational boundary. To understand aging effects on components and systems is to identify age-related degradation sites, understand aging mechanisms, and determine their root causes.

The process for determining when to maintain and how to maintain, the identification of risk significant and prioritized systems, structures, and components is *an integral part of managing aging in nuclear power* plants. Managing the influence and effects of aging requires knowledge of the aging and degradation processes described above, specifications (measures) of performance parameters and interpretation and analysis of functional indicators (measures), utilization of detection and monitoring methods, trending and recordkeeping, maintainence-refurbishment-replacement, and assuring the operational readiness of aged safety-related systems.

The mitigation of the effects of component and system aging on plant safety and the extension of plant life cannot be achieved through regulation by the NRC. Ultimately, it is the plant operator's responsibility to ensure continued safe operation of its plants. To do this, one must understand the aging and degradation processes in plant safety-related systems, components, and structures; develop a program of surveillance, *monitoring, trending, recordkeeping and analysis to* mitigate the effects of aging; and then commit to implementing a rigorous maintenance program to ensure plant safety throughout its operational life.

Table 1[*]
Degradation Mechanisms and Where They are Operative

Degradation Mechanisms	Where Operative
Thermal Embrittlement (High Temperature)	Ferritic Stainless Steels, Cast Stainless Steels
Irradiation Embrittlement	All Reactor Components
Fatigue (Low and High Temperature)	Rotating Equipment Supports and Piping Attached to Large Components
Corrosion Fatigue (Low and High Temperature)	Thermal Mixing Regions Esp. Carbon and Alloy Steels
General Corrosion, Pitting, and Wastage (Low and High Temperature)	Crevices and Hideout Regions, Low and No Flow Components, Safety Injection Systems, Service Water Systems
ID SCC (Low and High Temperature)	Weld Vicinity in Components, Off Chemistry Conditions
Weld Related Cracking (Lack of Fusion, Hot Ductility, Ferrite Depletion, Crevice Formation) (High or Low Temperature)	Similar Metal Welds, Wrought Materials to Castings, Low Ferrite Filler Joints, Seam Welds
OD SCC (Chloride Related) (Low and High Temperature)	Components Near Valves (Leaking) Insulation (Mirror), Seaside Plants
Hydrogen Embrittlement (Low Temperature)	High Strength, Low Alloy Components, Vessel Cladding (Ferrite Phase), Interface Between Vessel Cladding and Vessel, Anchor Bolts, Vessel and Pressurizer Supports
Erosion-Corrosion (High Temperature)	Steam Piping and Steam Separators, Heat Exchangers (i.e., Moisture Separator Reheaters), Turbine Blades
Low Temperature Sensitization (High Temperature)	Stainless Steel Components, Cast Components
Irradiation Assisted Stress Corrosion	Piping, Pressure Vessels
Binding and Wear	Components Within Pumps and Valves
Oxidation	Relay and Breaker Contacts, Lubricants, Insulating Materials Associated with Electrical Components
Thermal Runaway (Dielectric Materials)	Capacitors, Solid State Devices
Partial Discharges	Transformers, Inductors, Medium and High Voltage Equipment
Crevice Corrosion (Low and High Temperature)	Stagnant Regions, Weld Vicinity, Sleeved Regions, Welds with Backing Rings
Mechanical Wear, Fretting (Low and High Temperature)	Rotating Equipment
Creep and Swelling (High Temperature)	Vessel Internals (Radiation Assisted)
Microbial Induced Corrosion (Low Temperature)	Service Water, Heat Exchangers, Equipment Where Pressure Tests Performed, Equipment During Layup, Anchor Bolts, Diesel Generators
Dilution Zone Cracking (High or Low Temperature)	Dissimilar Metal Welds, Vessel to Clad Interface, Nozzle to Safe-Ends, Valves or Pumps to Pipe (Carbon Steel to Stainless Steel)
Insulation Embrittlement and Degradation	Cables, Motor Windings, Transformers

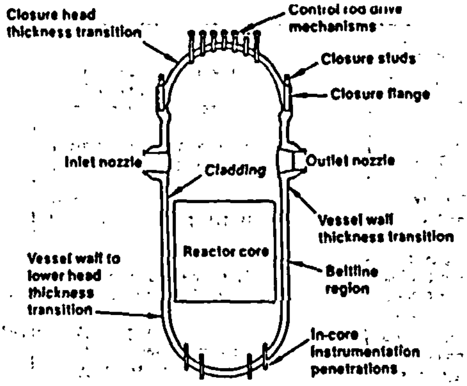
---

[*] Discussed at the ASME Section XI Special Working Group Meeting on Plant Life Extension

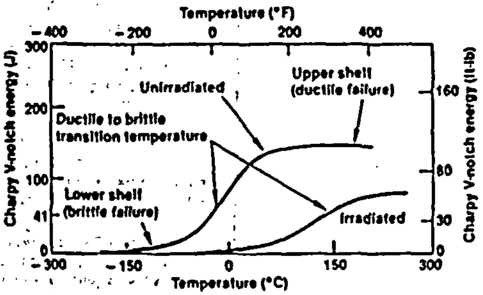
Figure 1

Understanding and managing aging of PWR reactor pressure vessels

Materials	Vessel	• Low alloy carbon steel - SA-533B-1, SA-508-2, SA-302B
	Cladding	• Austenitic stainless steel - Type 308 or 309
	Weldments	• Submerged arc (granular flux - linde 80, 91, 124 and 1092 manganese-molybdenum nickel filler wire) narrow gap submerged arc, shielded metal arc, and electroslag
	Closure studs	• SA-540 Gr. B24 Class 3
Stressors and Environment	Neutron flux and fluence, temperature, reactor coolant, cyclic thermal and mechanical loads, boric acid leakage	



Typical PWR vessel showing important degradation sites.



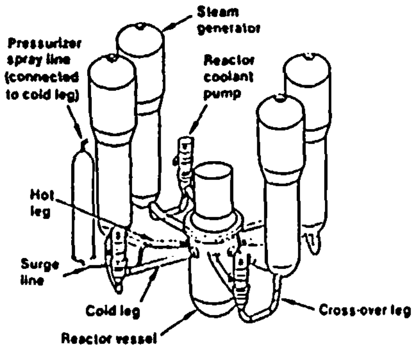
Effect of irradiation on the Charpy impact energy for a nuclear pressure vessel steel.

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Sites	Aging Concerns	Inservice Inspections, Surveillance, and Monitoring		Mitigation
		NRC Requirements	Recommendations	
Beltline region	Irradiation embrittlement – Chemical composition of vessel materials (Cu, Ni, P) – Drop in upper shelf energy – Shift in reference nil-ductility Transition temperature Environmental fatigue	Surveillance program to assess irradiation damage, i.e., shift in RT_{NDT} and drop in USE - 10 CFR 50 App. H Reg. Guide 1.99, Rev. 2 - acceptable PTS screening criteria - 10 CFR 50.61 Damage evaluation - 10 CFR 50 App. G Volumetric examination of one weld during each inspection interval -10 CFR 50.55a, IWB-2500 *Flaw detection and evaluation* - 10 CFR 50.55a, IWB-3000 Leakage and hydrostatic pressure tests 10 CFR 50.55a, IWA-5000	*Include fracture toughness and tensile* test specimens in surveillance program Develop use of reconstituted and miniature specimens Accelerated irradiation of reconstituted specimens Revise Reg. Guide 1.99, Rev. 2 to account *for phosphorus with low copper* Perform volumetric examination of all welds during each inspection interval Use state-of-the-art NDE techniques for improved reliability of defect detection, *sizing, and characterization* Use fatigue crack growth curves ASME SC XI, Appendix A Develop acoustic emission monitoring to detect crack growth	Flux reduction Inservice annealing ASTM E 509-86 Determine effects of *annealing and* reembrittlement rate
Outlet/inlet nozzles	Environmental fatigue Irradiation embrittlement Function of nozzle elevation (Potential impact of Reg. Guide 1.99, Rev. 2)	Volumetric examination of 100% *nozzle-to-vessel* welds and nozzle inside radius section during each inspection interval -IWB-2500 Volumetric and surface examination of all *dissimilar metal welds during each inspection* interval -IWB-2500	Use on-line fatigue monitoring Evaluate irradiation embrittlement damage	
Instrumentation nozzles CRDM housing nozzles	Environmental fatigue	Visual examination of external weld surface of 25% nozzles during system hydrostatic test -IWB-2500		
Flange closure studs	Environmental fatigue Boric acid corrosion (if leakage occurs)	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval-IWB-2500		

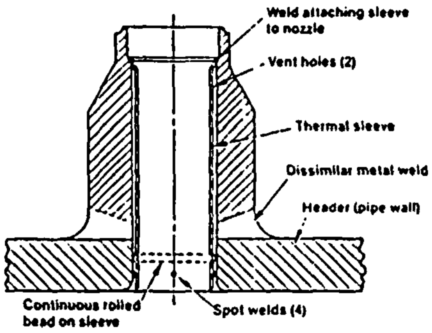
Figure 2

Understanding and managing aging of PWR RCS piping and nozzles

Materials	Main coolant pipe	- Centrifugally cast SS-Gr. CF8A and CF8M (W), Type 304SS and 316SS (early W plants), SA 516 Gr. 70 (CE), SA 106 Gr. C (B&W)
	Fittings	- Statically cast SS - Gr. CF8A and CF8M (W); SA 516 Gr. 70, Type 309L SS (CE, B&W); Type 308L SS (B&W)
	Cladding	- Type 308L SS (CE), Type 304L SS (B&W)
	Surge line	- Type 316 SS, cast SS - Gr. CF8M (some CE plants)
	Spray line	- Type 316 SS
	Nozzles on main coolant pipe	- SA 105 Gr. 2 (CE), Type 304N SS (W)
	Thermal sleeve	- Inconel SB-168
Stressors and Environment	Operational transients, temperature, flow induced vibrations, stratified flows, thermal striping, and thermal shocks	



Typical RCS piping for W 4-loop plant



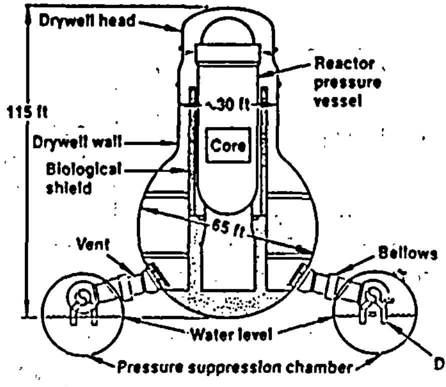
Typical nozzle-thermal sleeve assembly for a W plant

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Sites	**Aging Concerns**	**Inservice Inspections, Surveillance, and Monitoring**		**Mitigation**
Nozzles and thermal sleeves Charging Safety injection Surge Spray	Low and high-cycle thermal and mechanical fatigue	NRC requirements Volumetric and surface examination of 25% of butt welds including the following welds during each inspection interval 10 CFR 50.55a, IWB-2500: • All dissimilar metal welds • All welds having cumulative usage factor equal to or greater than 0.4 • All welds having stress intensity range of 2.4 Sm Same welds are required to be inspected during each inspection interval Flaw detection and evaluation - 10 CFR 50.55a, IWB-3000 Leakage and hydrostatic pressure tests 10 CFR 50.55a, IWA-5000 Cycle counting of specified design transients Tech. Spec. requirement	Recommendations Perform more frequent examination of nozzle welds having high cumulative usage factor Determine fatigue damage by on-line monitoring of coolant and piping temperatures, pressures, and flow rates in nozzles and horizontal portions of piping during operational transients, stratified flows, and thermal shocks Perform nondestructive examinations and loose parts monitoring to assess status of thermal sleeves Develop use of acoustic emission method to detect crack growth in the base metal and welds Develop techniques to monitor actual degree of thermal embrittlement in cast stainless steel piping: - Analytical modelling of inservice degradation - Metallurgical evaluation to characterize microstructure - NDE to establish correlation between ultrasonic attenuation and fracture toughness Monitor valve leakage in safety injection pipe Develop UT to detect flaws in cast stainless steel piping	Maintain full flow in spray line and operate it continuously to prevent stratified flow and thermal shock conditions Replace horizontal section of spray line with sloped section to prevent stratified flow condition Redesign piping to eliminate valve leakage
Terminal end dissimilar metal welds (between carbon steel components and stainless steel piping)	Low-cycle thermal and mechanical fatigue			
Surge line Spray line	Low and high-cycle thermal and mechanical fatigue			
Cast stainless steel piping Hot leg Cross-over leg Cold leg Fittings Surge line	Thermal embrittlement			

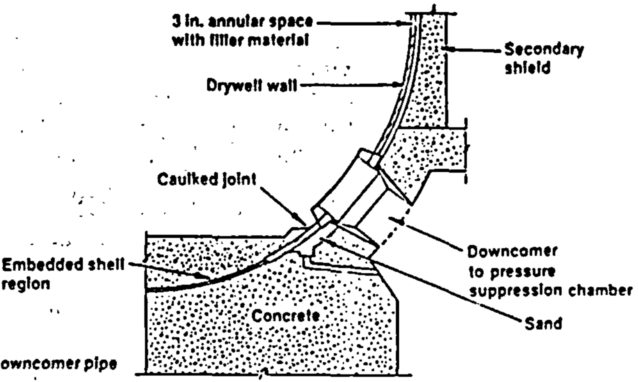
Figure 3

Understanding and managing aging of BWR Mark I containments

Materials	**Shell**	• Carbon steel - SA-516 Gr.70, SA-212 Gr. B
	Bellows	• Type 304 Stainless Steel
	Coatings	• Zinc rich, red lead and epoxy
Stressors and Environment		Corrosive internal environment, temperature, humidity, oxygen content, degraded fill material, moisture, microorganisms, cyclic thermal loading, leak tests



BWR Mark I type metal containment



BWR Mark I drywell base, concrete shield wall and sand pocket.

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)				MANAGING AGING	
Sites		Aging Concerns		Inservice Inspections and Monitoring	Maintenance
Drywell	Exterior surface near sand pocket (unsealed gap)	Aqueous corrosion and microbial influenced corrosion	Thermal, mechanical, and environmental fatigue	NRC Requirements Leak tests - 10 CFR 50 App. J *Recommendations* ASME subsection IWE • 25% visual examination of pressure retaining welds, and *coated and uncoated surfaces* during each inspection interval • Being reviewed by NRC to include in federal regulations Visual examination of caulked joints at embedded regions Boroscopic examination of exterior surface near penetration Wall thickness measurements Surface examination of bellows Monitoring coolant leaking from faulty bellows	Recommendations Maintain surface coatings Check bellows alignment Maintain caulked joints at embedment region
	Exterior surface (degraded fill material present)	Pitting and crevice corrosion			
	Embedded shell region (deteriorated caulked joint *at concrete-metal interface*)	Pitting and crevice corrosion			
	Pipe penetrations, Vent pipes	Galvanic corrosion			
	Exterior and interior surfaces (deteriorated coating)	Uniform attack			
Pressure Suppression Chamber	Interior surface (deteriorated coating)	Pitting	Thermal, mechanical, and environmental fatigue		
	Near waterline	Differential aeration			
	Below waterline	Microbial influenced corrosion			
Bellows	Heat-affected zone	Intergranular stress corosion *cracking*	Thermal and *mechanical fatigue*		
	Cold-rolled portion	Transgranular stress corrosion cracking			

Figure 4

Understanding and Managing Aging of Diesel Generators

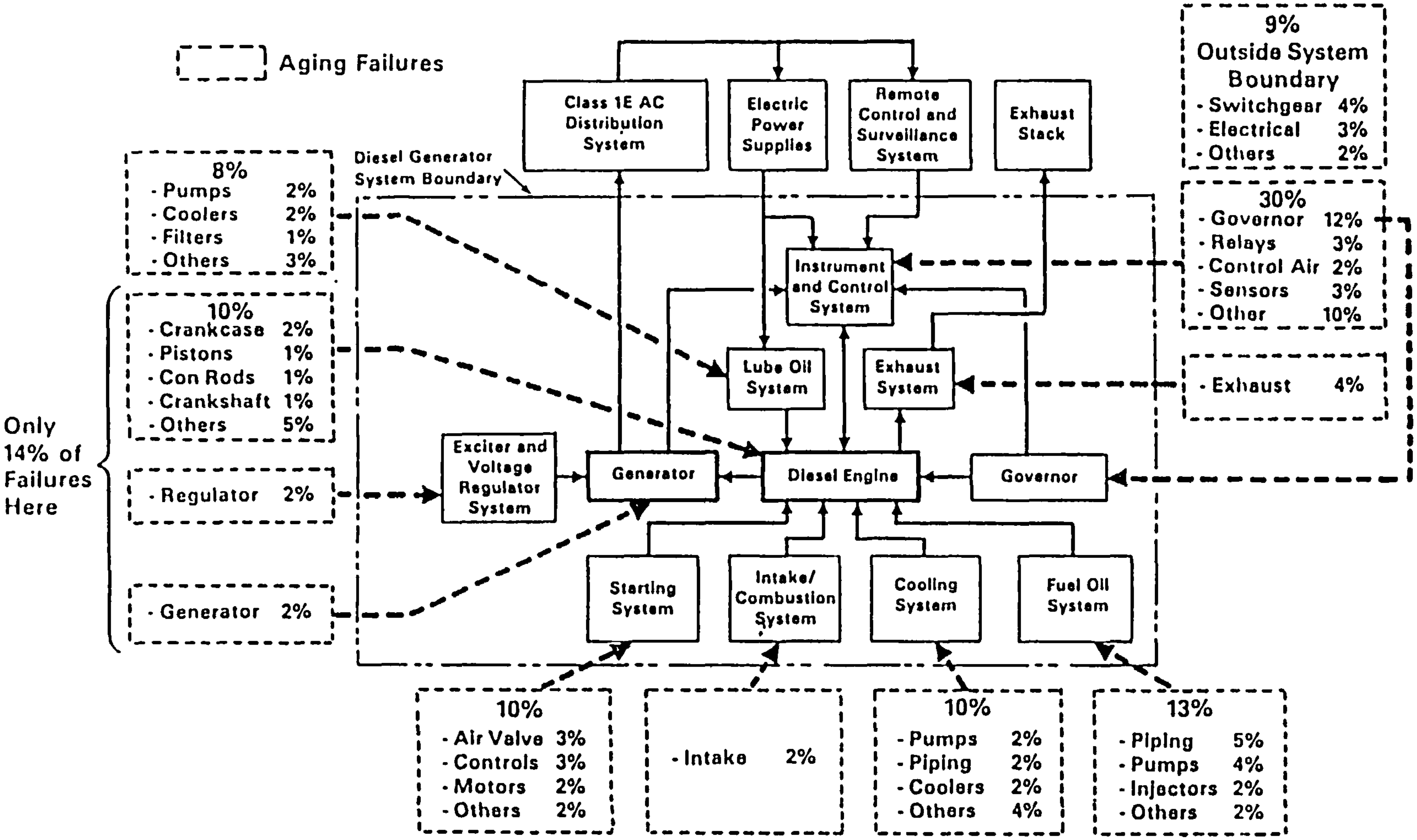


Figure 5A

UNDERSTANDING AND MANAGING AGING OF MOTOR-OPERATED VALVES

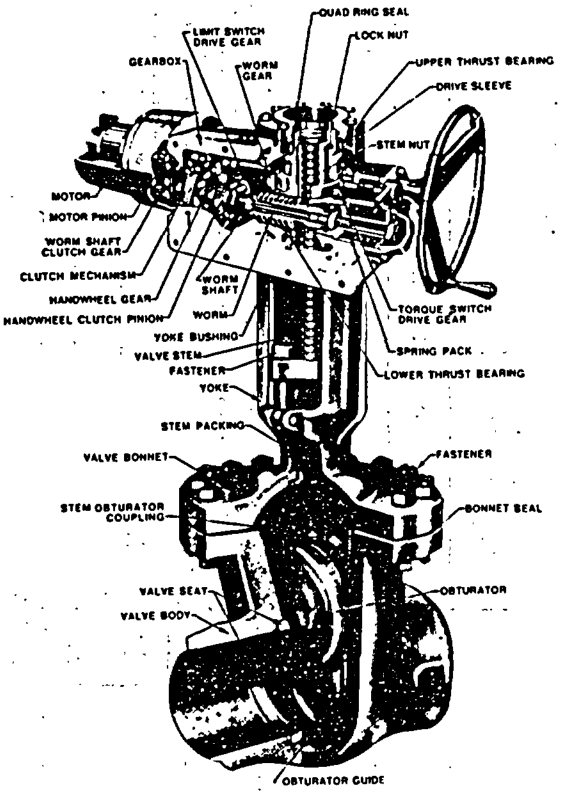


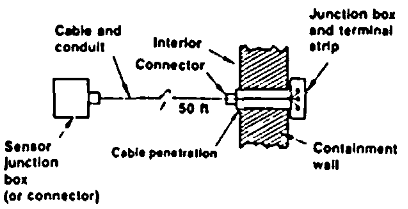
Figure 5B

UNDERSTANDING AGING (Materials, Stressors, and Environment Interactions)		MANAGING AGING	
Sites	**Aging Concerns**	**Inservice Inspection and Monitoring**	**Mitigation**
Motor Operator			
Gearbox assembly	Gear wear Shaft wear, distortion Fastener loosening Stem nut wear Stem lock nut loosening Spring pack response change Drive sleeve wear Clutch mechanism wear Seal wear,deterioration Bearing wear, corrosion Lubricant degradation, hardening	<u>NRC Requirements</u> ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV A. Verify valve obturator position B. Measure stroke time C. Measure seat leakage	Implementation of improved condition monitoring methods, predictive and corrective maintenance programs, and replacement strategies
Electric motor assembly	Bearing wear, corrosion Insulation (electrical) breakdown	IE Bulletin 85-03	
Switches	Contact pitting, corrosion Gear/cam wear Insulation (electrical) breakdown Fastener loosening Grease hardening (limit switch)	Requests that utilities develop a program to ensure that MOV switch settings are selected, set, and maintained correctly to accomodate normal and abnormal events	
Valve (Gate valve)	Obturator wear, corrosion Obturator guide wear, corrosion Yoke bushing wear Valve stem wear, distortion Fastener loosening Valve seat wear, corrosion Bonnet seal deterioration Stem packing wear, deterioration		

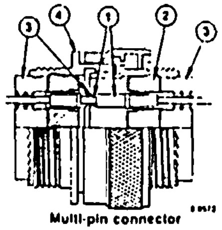
Figure 6

Understanding and managing aging of instrumentation cables, connections and penetrations for light water reactors

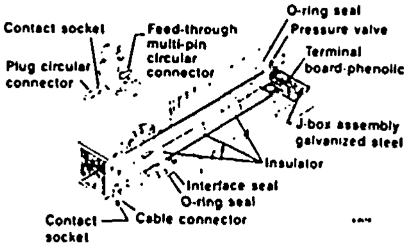
Stressors and Environment
- Temperature
- Radiation
- Oxygen
- Humidity
- Mechanical stress



Instrumentation cable system



Multi-pin connector



Typical electrical penetration

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)				MANAGING AGING		
Component	Subcomponents	Typical Material	Aging Concern	Inspection and Monitoring		Maintenance
				Preservice	Inservice	
Cable	Insulation	Crosslinked polyethylene	Thermal and radiation embrittlement, oxidation, cracking, and moisture intrusion	• Inspected when installed to verify qualification and installation requirements which include: 10 CFR 50.49 Calls for artifical or natural aging prior to E.Q. testing Reg. Guide 1.89 Qualified life may be demonstrated based on arrhenius theory and a surveillance and maintenance program IEEE-383 Provides industry guidance for qualifying Class 1E-cables and connections IEEE-317 Covers design, installation, and testing of electrical penetrations in containment structures	• No requirement for inservice inspection • Inspection after maintenance on a connection • Monitor measurement channels for indication of signal degradation or discrepancies between redundant channels during operation • During refueling outage end-to-end system tests include cables and connections Recommendations • Develop inservice surveillance criteria • Perform periodic inspections • Perform temperature and radiation mapping in containment cable locations	• Typically cable maintenance is done only when system performance has degraded or specific problems have been identified by plant operations or by system testing • Replace components at end of qualified life • Maintenance records will support licence renewal Recommendations • Develop advanced remote monitoring technology • Periodically replace cable in high temperature and radiation areas
	Jacket	Chlorosulfonated polyethylene	Mechanical stress, thermal and radiation embrittlement, and cracking, and moisture intrusion			
	Conductor & Shields	Stranded copper	Fatigue or corrosion			
Penetration	O-ring seal	Elastomer	Pressure leak, cracking			
	Contact socket	Gold plated copper	Wear with use			
	Interfacial seal	Dow corning sylgard	Cracking			
	Insulator	Polysolfone	Cracking			
Multi-pin connector	Pins and sockets	Gold plated copper	Wear with use Gold-solder chemical reaction			
	Inserts (insulation)	Thermal plastic polymer	Embrittlement, wear			
	Seals and grommets	Fluorosilicone elastomers	Cracking			
	Shells and rings	Aluminum or stainless steel	No aging concern			
Terminal strips	Terminal board	Glass filled phenolic	Embrittlement			
	Cable clamp lug and screw	Stainless steel	Broken or loose screws or dirty connection			
	Shrink tubing	Polyolefin	Cracking			
Junction box	Seals	Elastomer	Embrittlement			

PROPOSED IAEA PROGRAMME ON SAFETY ASPECTS OF NUCLEAR POWER PLANT AGEING AND LIFE EXTENSION

by

J. Pachner,* Atomic Energy Control Board, Canada
S. Novak,* International Atomic Energy Agency

This paper provides information on the proposed IAEA programme on safety aspects of nuclear power plant (NPP) ageing and NPP life extension, as recommended in June 1988 by the Advisory Group organized by the IAEA.

Background

At present, some NPPs are approaching or are even beyond the end of their originally planned service life. In the year 2000, approximately 160 plants will be 25 years old and 69 will be 30 years or older. Because of the potential impact of ageing degradation of NPP components, systems and structures on plant safety and reliability, many countries are devoting a significant effort to better understanding and managing the effects of ageing. The amount of work to be done is very large and, therefore, an exchange of information and cooperative work among IAEA Member States would be beneficial to all.

Recognizing the potential impact of ageing on plant safety, and in response to requests of its Member States, the IAEA commenced activities concerned with the safety aspects of NPP ageing three years ago. In 1985, it convened a Working Group of four consultants who drafted a report on safety aspects of NPP ageing to stimulate relevant activities in the Member States. At a Technical Committee meeting, organized by the IAEA in September 1986, 23 participants representing eleven Member States, the Nuclear Energy Agency of the OECD and the IAEA exchanged information, and reviewed and commented on the Working Group's report. In the summer of 1987, an IAEA Symposium on Safety Aspects of the Ageing and Maintenance of Nuclear Power Plants, held in Vienna, drew 140 participants from 28 countries and four international organizations. The Working Group met again in Vienna in the fall of 1987, and using the input from the Technical Committee meeting and the symposium, revised the first draft of the state-of-the-art report on safety aspects of NPP ageing. Finally, earlier this year, the IAEA organized an Advisory Group to explore further international cooperation on safety aspects of NPP ageing and NPP life extension. Participants at the Advisory Group meeting came from Canada, Czechoslovakia, France, Japan, Switzerland, United Kingdom, USA and IAEA; representatives of USSR and FRG were not able to attend.

After exchanging information on respective national experiences and programs, and discussing the IAEA secretariat's proposal, the group made the recommendations to the IAEA described in the following sections. It is envisaged that the proposed programme will be sponsored by the IAEA's Divisions of Nuclear Safety and Nuclear Power, with the Division of Nuclear Safety having the leading role in its implementation.

In this paper, the meaning of the term 'components' includes also plant systems and structures; 'ageing' is used in the broad sense and includes service wear; 'life extension' is used also in the broad sense and includes life assurance; and 'failure' is considered to occur when a component is unable to meet its minimum functional requirements.

Proposed IAEA Programme

The Advisory Group recommended to the IAEA a multi-year programme, Nuclear Power Plant Ageing: Safety Aspects and Life Extension, with the following mission statement.

To facilitate the exchange of information and to promote cooperation among IAEA Member States towards understanding and managing the ageing degradation of NPP components, with the aim of maintaining safe and reliable plant operation.

Two objectives were derived from the mission statement.

To establish and maintain, under the auspices of the IAEA, a programme of international cooperation for:

- increased awareness and understanding of ageing degradation processes,
- development of methods and guidelines to manage ageing for safe and reliable operation of NPPs.

The Advisory Group then designed a programme to achieve these objectives.

The programme is structured into four subject areas:

- review of operating experience and information exchange
- understanding ageing
- managing ageing - technical issues
- developing guidelines for managing ageing

Each of the subject areas is described in some detail below.

1. Review of Operating Experience and Information Exchange

1.1 State-of-the-Art Reports on Safety Aspects of NPP Ageing. The first report prepared by the IAEA will be published in the fall of 1988. It is intended to provide basic information on safety aspects of NPP ageing and a systematic approach to understanding and managing the ageing of NPP components. It was recommended that follow-up reports be periodically prepared to update the first report, and to provide more detailed information on selected materials and component types.

1.2 Exchange of Information on Ageing Related Failures/Events. The Advisory Group further recommended that Member States exchange information on age-related component failures and operational

---

* Mr. Pachner was chairman and Mr. Novak scientific secretary of the IAEA Advisory Group on Further International Cooperation on Safety Aspects of Nuclear Power Plant Ageing and Life Extension, 30 May - 2 June 1988.

events. The root causes of these failures and events, and their effects, such as system/plant unavailability, common mode failure, or reduction in defence-in-depth and reliability, should be provided for meaningful information exchange.

To facilitate the information exchange, Member States would be requested to examine their existing operational data, maintenance, testing and inspection records, and to consider how best to present information on ageing degradation in a form suitable for international information exchange. The format of this information should be in accordance with the IAEA guidelines that would be developed as one of the first activities of the programme.

1.3 Exchange of Information on Results of Studies on Understanding and Managing Ageing of NPP Components and on Plant Life Extension. It was recommended that Member States exchange information on their approach to, and the present status of their studies on NPP ageing and life extension.

2. Understanding Ageing

It is generally agreed that ageing of plant components cannot be stopped, but in many instances can be slowed down by appropriate preventive actions based on a thorough understanding of relevant ageing processes. Ageing is a complex phenomenon because of the many degradation mechanisms that exist, such as creep, fatigue, irradiation embrittlement, stress corrosion cracking, corrosion erosion, oxidation, wear (e.g. fretting). A programme of studies designed to achieve a better understanding of the ageing degradation of NPP components is outlined below. The studies are subdivided into five sections.

2.1 Analysis of Operating Experience. It was recommended under Section 1.1 above that operating experience of Member States' power plants be reviewed to identify age-related component failures and operational events. To help understand the ageing phenomena, the Advisory Group further recommended that these failures and events be analyzed in detail to identify and characterize the significant age-related degradation mechanisms. The obtained information would be useful in the evaluation of the performance of similar components in other plants, and also provide direction for follow-up studies.

2.2 Selection of Critical Components. It was recommended that a selection of critical NPP components be made for detailed engineering studies. The selection would be based, in the short term, on Member States' experience and recommendations, and, in the long term, augmented by probabilistic safety assessment (PSA) methodology. Other considerations in the selection of critical components should include potential impact of ageing on NPP reliability and economics.

2.3 Research & Development (R & D) - Laboratory Investigations. It was recommended that cooperative R & D studies be established to resolve technical issues and to fill knowledge gaps related to the ageing degradation of selected critical components. Some examples of recommended R & D are given below:

a. understanding the interaction effects among material, environmental and operational stressors to characterize age-related degradation mechanism such as stress corrosion cracking, corrosion erosion, irradiation embrittlement;

b. understanding the degradation of electrical equipment associated with radiation, temperature, humidity, length of service, and their interactions;

c. development and validation of remaining service life prediction models for mechanical and electrical components, and civil structures;

d. validation of accelerated ageing techniques used in the equipment qualification work.

2.4 Field Assessment of Ageing Degradation. It was recommended that the laboratory R & D described above be supplemented with field data related to actual service conditions, in-situ monitoring, etc.

2.5 Post-Service Examination and Tests. It was recommended that an evaluation be conducted of the ageing of pertinent components and material samples obtained from operating plants and from decommissioned plants.

3. Managing Ageing-Technical Issues

An understanding of ageing provides the key to the effective management of ageing processes needed to maintain safe and reliable operation of NPPs throughout their entire service life. Thus the studies on understanding ageing proposed above are closely linked to the studies on managing ageing recommended below. These studies are subdivided into three sections.

3.1 Analyses. The Advisory Group recommended that appropriate analyses be conducted to identify indicators capable of revealing component degradation before failure, and to relate indicator values to component functional capability.

3.2 Detection. Further, the development of practical detection techniques, capable of measuring and evaluating the indicators determined under 3.1 and thus detecting ageing degradation of the selected critical components before failure, was recommended. Evaluations should utilize absolute and relative comparisons of observed indicator values to determine present and projected component conditions.

3.3 Mitigation. It was recommended that effective methods for mitigation of ageing degradation of the selected components be developed. These methods include maintenance, refurbishment, replacement, operational changes and design improvements. In addition, predictive criteria to decide what and when preventive actions are required should be developed.

4. Developing Guidelines for Managing Ageing

The Advisory Group recommended that the technical information obtained and developed under areas 1, 2 and 3 of the programme be used as a basis for the development of the following guidelines for managing NPP ageing.

4.1 Generic and component-specific guidelines (for the selected critical components) for operational and maintenance data collection, records and documentation, which are needed for evaluation of the contribution of ageing effects to component degradation.

4.2 Component-specific guidelines for monitoring ageing degradation. These guidelines should include:

- parameters to be measured

- practical monitoring techniques
- frequency of monitoring (testing, inspection, on-line monitoring etc., as applicable)
- acceptance criteria

4.3 Generic as well as component specific guidelines for maintenance methods and practices that are effective in mitigating ageing degradation. These guidelines should address:

- maintenance
- refurbishment
- replacement

4.4 Guidelines for operating practices or suggested operating conditions that could reduce excessive age-related degradation. For example, the guidelines should address operational transients associated with start-ups, shutdowns, load following, etc.

4.5 Guidelines for the revision of applicable IAEA codes and guides to include explicitly ageing and life extension issues.

4.6 Guidelines for more effective management of ageing in the design of future plants.

Action Plan

To implement the proposed programme, the Advisory Group devised an action plan which is described below. The success of the programme will depend to a large degree on the support and the resources provided by the IAEA, the Member States and other organizations, such as utilities and research institutions. Obviously, Member States will contribute according to their own situation and needs. Any contribution will be valuable and should be encouraged.

Two planning principles were established at the outset of the planning process:

- ensure the cooperation and coordination of the programme with relevant programmes of the IAEA and other international organizations to prevent unnecessary duplication of effort, and to achieve greater effectiveness;

- subdivide the action plan into short-term (1989-90) and long-term plans.

The recommended action plan is given below. Where applicable, the action items are cross-referenced to the programme items given above.

Short-Term Plan (1989-1990)

1. The IAEA to seek and obtain support for the programme from Member States.

2. Publish the State-of-the-Art Report on Safety Aspects of NPP Ageing in the fall of 1988 (re: 1.1).

3. Set up the International Working Group (IWG), which will also steer the programme.

4. Select critical components on the basis of Member States' experience and recommendations, and initiate pilot studies on ageing of a limited number of components (re: 2.2).

5. Prepare guidelines for records keeping (re: 1.2, 2.3, 2.5, 3.2, 4.1).

6. Set up procedures for international information exchange (re: 1.2, 1.3, 4.1).

7. Organize topical meetings according to the IWG recommendation and widely distribute proceedings (re: 1.2, 1.3, 2.3).

8. Consider initiation of coordinated research studies (re: 2.3).

Long-Term Plan (After 1990)

IWG to assess results of the short-term programme and to recommend a long-term plan consistent with current needs and available resources.

The following subjects should be considered:

- operating experience review and international information exchange (re: 1.2 and 1.3)
- coordinated research and development studies (re: 2.3 and Sec. 3)
- risk significance of ageing phenomena and prioritization of actions using PSA techniques (re: 2.2)
- in-depth analysis of operating failures/events (re: 2.1, 2.3, 2.4, 2.5 and sec. 3)
- methods to assess residual life (re: 2.3 and 3.2)
- detection and monitoring methods (re: 3.1 and 3.2)
- predictive criteria for ageing mitigation (re: 3.2 and 3.3)
- maintenance methods (re: 3.3)
- guidelines for managing ageing (re: 4.2, 4.3, 4.5, 4.6)

Conclusion

It is felt that the above outlined programme, coordinated by the IAEA, would contribute significantly to maintaining adequate safety and reliability of the ageing plants both during and beyond their originally planned service life.

Acknowledgements

The programme proposal described above is a result of the efforts of and the spirit of co-operation among the following participants at the Advisory Group Meeting: J. Chadha of Canada, S. Kacmary of Czechoslovakia, J.Y. Henry and B. Magnon of France, S. Ueda of Japan, H. Boyne and R. Warner of Switzerland, J.W. Seddon of United Kingdom, J.P. Vora of U.S.A. and P. Dastidar, V. Lysakov, G. Woite and M. Cullingford of IAEA.

SAFETY - AGING - LIFE EXTENSION
A LONG TERM VISION
FROM THE ENGINEERING LEVEL

Emil Bachofner/Ulf Sjöö

Summary

Much has been said in the nuclear business about aging, life extension, requalification etc. These items will be even more relevant when an increasing number of plants in the world reaches the age of 20 years or more.

It is not unique that technically complicated units are getting old but must be operated at a high level of availability and reliability. Lessons can be learned from the heavy aviation industry with its history ranging from the thirties.

The overall safety of a nuclear plant is defined not only by the structural safety of the main components but to a great extent by all the components as represented in a plant specific probabilistic safety assessment (PSA). The contribution from each component and system can be determined and quantified. In contrary to the start up phase of the plant, criteria for requalification are not fully established.

Not all components in the plant are expected to have a lifetime corresponding to the reactor vessel. It is therefore of great importance that individual criteria for components and systems are established and that maintenance and surveillance is applied to this. This has been done for a number of decades in the aviation industry.

In this paper we try to define research and development or, as we call it, ordinary engineering work that must be done in order to maintain the plants on their original safety levels. We also try to identify relations between parameters used for components used in PSAs and for components in the field work.

Relations between safety and component function

Safety can be defined in different ways. In nuclear reactors safety means to prevent release of radioactive materials which is more or less the same as to prevent core damage. In our reactors core damage must be avoided by active system function such as reactor shut-down, residual heat removal or emergency core cooling. System performance is therefore necessary.

The reliability of a system is depending on the system structure and the reliability of involved components, as well as how often the system function is needed. Thus reliability criteria for a reactor shut-down system is different from that of an emergency core cooling system. With modern methods for reliability analyses the criteria for system function can be specified individually if the safety level for the whole plant is set. This is also the case for different components within the system. The problem is that reliability data are needed for all components if the reliability for the whole system shall be calculated. In the first PSAs there were extended discussions about the possibility of finding such data for the components. This is now history. From our point of view, it is essential to show that the reliability for key components corresponds with the assumptions used in the PSA for each unit. Here some questions must be asked:

- What constitutes a safety level for a unit and how is it set?
- What components are key components and how are they selected?
- How to define reliability data for components?
- How can the reliability for components be verified?

In this paper we may not be able to answer all these questions, but we can make an attempt to identify necessary work to make it possible to verify safety of aging units. For the time being we want to mention one important parameter which describes the safety level for a unit, that is the calculated frequency for core damage. A core damage must not necessarily be a core melt to cause a lot of trouble for the utility, a substantial release of activity even only inside the containment is enough. Years of shut-down for at least the involved unit will lead to substantial economic consequences.

If one accepts US safety goals, the tolerated frequency for core damage is 10^{-4}/reactor year. In Sweden with 12 reactors limited to 25 years of operation this means a probability for one core damage for the period ranging from 3 to 50 % depending on uncertainty bands. The question is if this is acceptable out of economic or insurance points of view. Our personal point of opinion is that the Swedish and the global safety level must be at a much lower frequency of core damage, somewhere in the area of 10^{-6}/reactor year. The task for the nuclear power utilities must be to verify these safety levels, and then the component reliability will come into focus.

Safety criteria and component function

In the Technical Specifications (Tec Spec) for Swedish nuclear power plants systems or parts of systems are defined as "Equipment of importance for safety." Principally all components included have safety criteria attached. Tec Spec is a rather square tool because they assume that every component has a reliability comparable with new ones. For all components which can be tested under realistic conditions the problem is limited to reveal an ongoing degradation. But how is the situation for all other equipment?

The result from PSA here offers clear guidance because it takes care of both the reliability structure of the system and numerical values for reliability. In plant specific PSA's estimations have been made for probability of component response at demand, probability for failure during system action and environmental conditions for the component. Critical components can easily be found from PSA, as well as the requirements for those components. Generally the following systems are governing the core damage frequency in our BWR's.

- Reactor pressure control system (relief valves)
- Systems for water supply to the reactor pressure vessel (Feedwater system, aux. feedwater system, emergency core cooling system)

- Systems for residual heat removal (Primary and secondary cooling systems)
- Systems for reactivity control (Scram system, Control rod manoeuvring system, Boron system)

Not all components in these systems are equally important because redundancy and diversification are different. However, as an example, the electric operated relief valves are found to be "key components."

Verification of component performance

During design and commissioning of new power plants many component tests are made. Typical tests are performance of components under realistic environmental conditions. Unfortunately these tests are expensive and time consuming, and are therefore kept to a minimum. A higher number of qualification tests are normally executed in new units.

Once complete qualification tests are executed, new questions appear concerning life time (qualification time), requalification or exchange of new (qualified) components. Here it must be made clear that good maintenance not automatically guarantees sound components.

The problem is that components can degrade without any visible signs. Insulation material degrades and plastic details change in shape. Only a complete test procedure with realistic conditions can reveal insufficient performance. A typical question to be answered is for how long time a pump motor in the cooling system for the suppression pool will operate at the age of 15 years at a temperature in the suppression pool of 95°C. Related to the safety analyses the pump must be able to perform for several weeks to fulfill the required safety level. The conclusion is that we have to answer the question.

Definition of used terms

We must define certain terms and connect them to both the PSA and the field work as follows.

- Environment, environmental factors and environmental qualification
- Aging - life time
- Methods of testing
- Tolerance criteria

As this field is large, it is impossible to be complete in a paper like this. The demand from the safety point of view is that work has to be done to give necessary knowledge about key components.

What requires safety in old units?

How can we gain a high safety level in aging plants? The same problem is treated in the heavy aviation business in a very stringent way. All components, from engines to the carpet in the cabin, are certified. Allowed time for different components in the aircraft is limited both concerning calendar time and operation time. Tolerance criteria are set and absolutely clear, as well as procedures for requalification.

We must be very sure about the following aspects:

- Documentation of original qualification.
- Criteria for setting life time and documentation of life time.
- Estimation of remaining life time.
- Procedures for requalification.
- Procedures for purchasing replacement components.

The last issue has to be recognized because old components often are replaced by new with completely new technics which requires a new qualification procedure.

Reliability technics and field work

The efforts to find connections between reliability parameters and component techniques must be intensified. Not all components must be "gold plated," but some must definitely be of high quality. Quality assurance-, maintenance- and safety departments must together establish a good technical and economic acceptable solution of the problem.

The goal for our work is to convince ourselves that all safety related equipment can respond to environmental challenges.

In our work we have identified some necessary steps in this work:

- Specify safety functions
- Specify criteria
- Application of existing standards
- Make quality control plans
- Testing of components
- Documentation of test result
- Maintenance program
- Program for requalification

All needed functions for which credit is taken in different scenarios must be mapped. As pointed out above the PSA is the main tool for this task. Based on the PSA we have to make detailed specifications that gives an envelope of criteria for each component. When the functions are specified and all criteria are formulated it is possible to find what standards can be applied, quality control plans can be written and qualification programs can be documented.

Once new components are installed it is necessary to have programs for periodic testing, surveillance testing and maintenance to ensure that the reliability is high enough through the whole life time.

We have in our organisation formulated a project treating the following questions:

- Do the periodic surveillance test procedures used today verify identified safety funcions?
- Are tolerance criteria given so decisions can be made about replacement or requalification, based on actual test results?
- Does test frequency correspond with safety analyses?
- How is experience from the tests taken in account to refine test procedures?

- Do we have a systematic program to analyze the test results, and experience from maintenance in order to identify beginning degradation?
- Are these experiences used to select good and bad equipment for further development in safety?

The questions above have to be answered through work in separate levels. This work will probably be carried out in different organisations such as research centres, laboratories and within the utility organisations.

How good are we today? What has to be done?

We believe that we in the nuclear business are qualified to evaluate our components. There is a broad experience from maintenance and design work and from most of the components which can be tested during realistic conditions.

However, it is clear that some components do not have necessary documented verification which prove the required performance during specified conditions. Further more, clearly specified limitation of life time is not always available for key components, not everything can last 40 years. Therefore it is essential to set these parameters, to document them and to live with them, not least from the economic point of view. It is not possible to bring up all the necessary work which have to be done and we would only like to point out the disciplines to involve.

- Research
- Development
- Common field and engineering work

UTILITY PERSPECTIVES ON NUCLEAR PLANT AGING

J. Thomas, D. Edwards, and G. Sliter

Summary. This paper presents utilities' perspectives on the subject of nuclear plant aging stemming from their review of about twenty reports prepared by U.S. National Laboratories under the U.S. NRC Nuclear Plant Aging Research (NPAR) program. To promote proper interpretation and use, definitions of aging and related terms are proposed which refine those appearing in the NPAR reports. Next, the way that utilities manage aging in operating plants and the role of aging in plant life assessment are discussed. The main conclusions of the paper are 1) the NPAR view of aging as an issue in itself is too diffuse to allow focus on the important but limited aging aspects appropriate to life assessment, maintenance and equipment qualification, 2) an overly broad interpretation of aging effects can be counterproductive in terms of safety and economics, and 3) consideration of changes in utility aging management programs should begin with an assessment of the effectiveness of existing practice and technology. Recommendations are offered concerning future activities of the NPAR program and utility interactions with the program.

Introduction

The authors of this paper agree with the NRC's theme for this *symposium*: understanding and managing aging are crucial to ensuring safety in nuclear plants. Also, they believe that there should be more industry consensus about the definition of aging, the manner in which aging impacts safety, and the extent to which the nuclear power industry already understands and manages aging. From a utility perspective, it appears that additional work and cooperation are needed to achieve consensus in these areas, especially with respect to the extensive NRC Nuclear Plant Aging Research (NPAR) program.[1]

Begun in 1983, the NPAR program has been addressing a comprehensive list of components, systems, and structures, with virtually all of the research being performed at national laboratories. Phase I of the NPAR studies consists of a preliminary identification of modes of aging degradation and interim recommendations for inspection, surveillance and monitoring (ISM) methods. Phase II of each study intends to use mainly tests on aged equipment to verify final recommendations for improved ISM methods, effective maintenance practices, and methods for evaluating residual life of components, systems, and structures. In view of their intended use in the regulatory process,[1] recommendations from NRC studies could have a significant impact on (1) utility operational and maintenance requirements during the remaining licensed term of plants and (2) requirements for plant license renewal.

Since 1984, the NRC has been discussing possible avenues of utility participation in the NPAR program. The main form of participation agreed upon initially was utility review of draft NPAR reports. Since 1986, the Electric Power Research Institute (EPRI) has coordinated technical reviews of about twenty NPAR reports. Initially, this was done by the EPRI Equipment Qualification Advisory Group (EQAG). Later, the Nuclear Utility Plant Life Extension (NUPLEX) Steering Committee participated as well. (The latter committee has recently evolved into the NUMARC NUPLEX Working Group of the Nuclear Management and Resources Council.) Some of the utility comments were used by NPAR researchers to revise reports with the intention of clarifying their technical content. The two-year review and comment effort has apparently been of mutual benefit to utilities and to the NRC and its NPAR contractors.

Utility comments on several of the reports pointed out that the definitions of aging and related terms in the NPAR reports could benefit from further refinement and some degree of consensus on the part of the NRC and the utility industry. In short, the understanding of aging should begin with clear, agreed-upon definitions of aging terminology.

Utility reviews also observed that, for the most part, the NPAR reports gave little acknowledgment of the fact that aging is not a newly discovered phenomenon in nuclear power technology. Aging was considered or "managed" either explicitly or implicitly in the design of plant components, systems and structures to the extent allowed by the state of technology when the plant was designed and licensed. Aging management continues today with the implementation of improved maintenance and ISM programs in the plants. In short, management of aging began with the design of plants and has continued during their operation. Consideration of further changes in utility aging management programs should begin with an assessment of the effectiveness of existing programs, practices and technology.

This paper proposes refinements in the definitions of aging terminology appearing in NPAR reports. The refined definitions may help to clarify the answers to such questions as: Do all types of aging lead to loss of function and impairment of safety? Should a failure due to an installation or maintenance error be classified as aging-related? Should we expect failures to increase as a plant enters the latter stages of its current licensed term? Is it appropriate or technically justifiable to construct mathematical functions of increasing rates of aging failures for use in probabilistic risk assessment models?

After revisiting the definition of aging and how it affects plant safety, the paper gives an account of the existing ways in which aging is managed in nuclear plants. Next it describes the role of aging in plant life assessment, in contrast to its role in maintenance or equipment qualification programs. Finally, recommendations are offered concerning future activities of the NPAR program and utility interaction with the program.

Aging Terminology and Definitions

The NPAR program's definition of aging is "the cumulative degradation that occurs with the passage of time in a component, system, or structure [which] can, if unchecked, lead to a loss of function and an impairment of safety."[1] Without further explanation or refinement, this definition can be misleading. For example, does it mean that all aging degradation, if unchecked, can lead to an impairment of safety? Does "checked" mean monitored, mitigated, or both? Actually, as discussed later, there are forms of aging that do not produce failures, do not impair safety, and therefore need not be checked, except perhaps for operational purposes. According to NPAR[1] this degradation takes place because of one or more of the following factors:

- material degradation from natural internal chemical or physical processes;
- stressors from improper storage, operating environment, or external environment;
- service wear;
- excessive testing; and
- improper installation, application, or maintenance.

We strongly disagree that human errors in installation and maintenance should always be classified as aging.

The refinements to the NPAR definition proposed below are intended to foster proper interpretation and use. Admittedly, no matter how much we redefine terms there will be room for various interpretations and misunderstandings. However, the authors believe that the effort will benefit all in the coming years of continuing dialog on aging.

The following definition of aging in nuclear power plants is proposed:

> **Aging is the net gradual degradation in the physical condition of a component, system or structure due to environment and service.**

Note that aging can produce either a degradation or an improvement in physical condition. Examples of improvements from aging are the increased ductility of steel components from self annealing under process heat and the increased strength of organics and concrete from continued curing during operation. Nevertheless, we propose retaining the term "degradation" because the term "aging" is customarily taken to be synonymous with "aging degradation." The term "net" rather than "cumulative" is used because the latter is more likely to be misinterpreted to mean that aging effects always increase monotonically and are irreversible. Use of the term "net" may also help to make clear the fact that repair or refurbishment can reduce or eliminate accumulated aging degradation.

Another element of this definition that warrants some discussion is the characterization of aging as "gradual." Gradual means small continuous or incremental changes such as would occur under normal operation, including expected transients (upset plant condition) producing relatively small, temporary increases in environmental conditions. The term gradual excludes from the definition of aging the relatively greater degradation produced by the sudden and short-term environmental and service extremes created by such occurences as loss of coolant accidents (emergency or faulted plant condition) and gross equipment operation or reassembly errors. This distinction between gradual and sudden changes in physical condition is consistent with the distinction between aging and accident degradation made in traditional equipment qualification practice and standards.[2] The definition of aging in the industry standard for qualifying electrical equipment is "the effects of operational, environments, and system conditions on equipment during a period of time up to, but not including design basis events." The definition of aging in an EPRI-sponsored review of equipment aging theory and technology is[3] "in-service deterioration due to environmental or operational stress."

Note also that the proposed definition of aging excludes the phrase "which can, if unchecked, lead to a loss of function and an impairment of safety," because not all aging degradation leads to loss of function and not all losses of function impair safety. As we will discuss below, some aging degradation is benign and need not be checked.

Other Aging-Related Terms

To further clarify aging concepts and their significance to safety, it is useful to define other terms related to aging. Such terms as "aging mechanisms," "aging stressors," "failure mechanisms,"failure modes," "failure cause," etc. are often used loosely or interchangeably in nuclear power literature. This problem has persisted within the extensive, multi-contractor NPAR program, as was pointed out in several of the utility comment papers.[4]

We will focus in this paper first on the terms "aging mechanisms," "aging degradation," and "aging stressors," then on terms related to failure.

> **Aging mechanisms are physical or chemical processes (such as wear, erosion, creep, corrosion, and oxidation) that result in aging degradation.**
>
> **Aging degradation is a debilitating change in physical properties (such as dimensions, ductility, fatigue capacity, and mechanical or dielectric strength).**

These terms are often mistakenly used interchangeably. Also, it is confusing when the term "aging mechanism" is used to mean other things such as "failure cause" and "failure mode." It is misleading as well to refer to "aging stressors" as "aging mechanisms."

> **Aging stressors are the environments (heat, radiation, humidity, reactive chemicals) and service conditions (operational cycling, electrical/ mechanical loads, vibration, testing) that induce aging mechanisms.**

It is important to distinguish between two classes of aging stressors -- "normal" and "abnormal" stressors.

> **Normal stressors are the actual environments and service conditions experienced by a component that has been properly fabricated, installed, operated, tested, and maintained.**

Normal stressors are design-basis stressors that should have been accounted for in design. They may not have been accounted for either because they were overlooked or because knowledge of their existence was beyond the state of the art. In these cases they are called "unanticipated" normal stressors. Thus normal stressors are either designed-for or unanticipated.

> **Abnormal stressors are environments and service conditions due to design or application errors, fabrication defects, and improper installation, operation, or maintenance (including excessive testing).**

Note that an environmental stressor that was overlooked in the design of a component is normal albeit unanticipated (for example, self-heating of a solenoid coil) whereas one that was created because of a design error would be abnormal (for example, the overheating of a coil that was sized too small). Normal stressors exist regardless of whether the design is performed properly, whereas abnormal stressors are the direct result of a design error.

The concept of "normal aging" was introduced in the NPAR report on check valves[5] and that of "abnormal aging" in the NPAR report on diesel generators.[6] These concepts do not appear in any other NPAR reports.

Failure-Related Concepts and their Safety Significance

Aging degradation from correctly-designed-for stressors is benign in that it is expected and does not cause failure within the specified design life of a component. Failures within the design life are caused only by unexpected aging degradation which includes both "unanticipated normal aging" and "abnormal aging" (aging degradation produced by abnormal stressors). Aging degradation that can lead to a failure is unacceptable in that a failure is costly and might lead to impairment of safety.

Our primary concern should be aging degradation and failures that impair safety. But not all aging degradation or failures impair safety; e.g., aging of non-safety-related components, systems, and structures (henceforth to be referred to collectively as simply "components") does not impair safety except in cases where failures may increase challenges to safety systems. Also not of safety concern is aging of safety-related components that does not have the potential for affecting their safety-related function. This applies to (1) expected degradation that does not fail safety-related components, (2) degradation of non-safety-related subcomponents or (3) degradation of safety-related subcomponents that does not play a role in its eventual postulated failure (for example, an EPRI research program[7] has demonstrated by test that thermal, radiation, and cyclic age conditioning does not reduce the seismic ruggedness of many types of small components or subcomponents in nuclear plant electrical equipment). This class of aging might be called "safety-significant" aging to distinguish it from "benign" aging which can not produce component failure (e.g., corrosion of a name plate) or which, even if it produces a failure (e.g., of a non-safety-related component or subcomponent), does not affect plant safety.

So far in our discussion we have tacitly implied that a failure of a safety-related component, whether or not caused by aging, is a direct threat to plant safety. This is not so. The plants have been designed with the basic philosophy of defense in depth and redundancy. Both of these concepts are cornerstones to the design of all safety systems and components. The first is particularly characteristic of passive components/structures that prevent release of radioactivity in the event of an accident. The second is particularly characteristic of active components/equipment that are deployed in redundant safety trains so that the ability to bring the plant to a safe shutdown condition is not compromised by a single failure. The objective of equipment qualification is to eliminate common mode failures of safety-related components in redundant safety trains. Utility reviewers observed that, in general, NPAR reports overstated the safety significance of aging mechanisms and failures in that these considerations were ignored.

An aging mechanism that leads to component failure is called a **failure mechanism**. Examples of aging-related failure mechanisms are corrosion and wear. But a failure mechanism can be non-aging-related. An example is intrusion of a foreign object large enough to block flow in a pipe.

Failure mode is defined as the manner of failure, such as the opening of a circuit due to corrosion or the seizure of a bearing due to wear. Failure mode has a somewhat different meaning in probabilistic risk assessment, referring to an action that a component fails to perform. For example, a seized bearing might produce the failure mode of insufficient pump head. Specifying failure mode is much more straightforward than identifying the basic cause of failure, as any person who has tried to fill out or interpret a failure report will attest. The stated cause can differ depending on how deep the evaluator reaches to find the "basic" or "root" cause. Historically, this has been a chronic and substantial problem for reliability data bases.

The NPAR program has taken a major step toward unraveling and systematizing the classification of failure causes in its aging failure survey conducted by Idaho National Engineering Laboratory.[8] This study adopted the following definition:

> **Failure cause is an underlying or initiating event or condition that produces the failure of a component.**

Although it would be difficult to improve, this definition opens the way for confusion. If a shaft fails by wear, is the failure caused by its worn <u>condition</u> or by the <u>event</u> in which a maintenance person installed a bushing upside down? Some clarification was provided by the comprehensive categorization scheme developed by the researchers. Their scheme is more detailed than that used in the Nuclear Plant Reliability Data System (NPRDS) and consists of three levels of increasing detail. The first (general) level is of most interest here since it encompasses many of the concepts discussed above. The five categories in the first level of failure causes are:

- design/manufacturing/construction/quality assurance inadequacy
- environmental stress
- human actions
- supervision/management inadequacy
- unclassifiable cause

Each failure category is further classified as aging-related or non-aging-related, depending on whether the effects are time-dependent or immediate.

Since the "environmental stress" category includes wear and other operational stresses, it is equivalent to what we have called "aging stressors." Many of the detailed cause categories other than those under "environmental stress" include the following statement: "Errors or inadequacies associated with these causes can cause accelerated aging. In order for a failure to be classified as aging-related when using one of these codes, the failure description must also contain an aging-related environmental stress effect or failure cause (described under the environmental codes) resulting from the error or inadequacy." Thus it turns out that a portion of the 31 percent of all failures reported by the NPAR survey as "aging-related" actually had more basic failure causes of errors and inadequacies in design, manufacturing, construction, human actions, etc. It is important to recognize that any errors and inadequacies included in the failure cause categories listed above do not increase with plant age. On the contrary, such failure causes predominate in the early years of plant operation and can be expected to continue to decrease as plant operators grow in experience and wisdom.

In the same vein, it is likely that for a portion of the causes categorized as environmental stress (and aging-related), the basic cause was actually produced by errors and inadequacies that may not have been identified. It is possible that a busy maintenance engineer might attribute mysterious failures to "aging."

It appears that additional effort must be made in improving both the failure data base and failure categorization schemes before the objectives of an aging failure survey can be met.

The main conclusion from this examination of definitions and terminology is that there is a crucial difference between characterizing a failure as aging-related (meaning simply that aging degradation played some role in the failure) or as caused by aging (meaning that unanticipated normal aging was the root failure cause).

Understanding Aging

The nuclear power industry is at the cutting edge of knowledge in the area of component aging. The need to account for aging of organic components in the qualification of electrical and mechanical equipment has led to detailed, sophisticated research into methods for simulating aging in the laboratory.[9] In the area of metal fatigue and nondestructive examination, nuclear power technology and practice is rivaled only by the aerospace industry. In knowledge of the properties of pressure vessels and piping materials, including radiation effects, the nuclear industry has no peer.

This understanding of aging has been increasing since the early days of commercial nuclear power and has been applied in plant design. Codes and standards have had a corresponding growth in both numbers and technological content. The understanding of course relates only to what we have defined as "normal aging." **No amount of understanding of "abnormal aging" can be applied in the design of a plant.** Abnormal aging is produced by errors and inadequacies which do not necessarily increase as plants age. They are best addressed by identification and prevention through good practice, and, if anything, they decrease as operational experience increases.

As with all technologies, our understanding of aging is not complete. Research can help, but the research should focus on uncovering unknown or not completely understood aging mechanisms, and on searching for improved, practical ISM techniques (good examples of this are the efforts by EPRI, NRC, and DOE to examine cable aging and condition monitoring).[10,11] In general, the NPAR program might benefit by putting more emphasis on using failure experience in other industries with old facilities (for example, early fossil fueled plants) in an attempt to uncover any aging mechanisms that may have yet to appear in nuclear plants. This approach proved to be useful in the NPAR work on diesel generators.[6]

Gaps in understanding are reduced not only by ongoing research but also by operating experience, which occasionally points out unanticipated aging mechanisms. Operating experience contributes to our knowledge of aging through analysis of failures and also (preferably) through inspection, surveillance, and monitoring prior to loss of component function. ISM checks both normal and abnormal aging degradation.

Managing Aging

Although state-of-the art design, design verification (qualification), and performance evaluation (ISM and failure evaluation) are vehicles for understanding aging, utilities use them also as tools for managing aging. These are supplemented with the important aging management tool of maintenance. Regularly scheduled preventive maintenance minimizes the rate of growth of some aging mechanisms and rejuvenates components through refurbishment and replacement of subcomponents prior to wearout. The NPAR reports could have given more credit to existing aging management programs in operating plants. This was particularly evident in a study[12] that attempted to construct mathematical models of aging for eventual use in risk evaluations. The study did not account for the effect of maintenance on reducing failure rates. (On the other hand, maintenance can also increase failure rates as explained below.)

With the concepts of aging and failure discussed above as background, we can now address whether the perception is correct that aging failures are bound to increase as a plant advances in age and that this warrants an across-the-board increase in maintenance and ISM.

Consider the classical "bathtub" curve that roughly characterizes the failure rate of a population of similar components (Fig. 1). The curve begins high due to "infant mortality" from gross design, fabrication, and installation errors. It then decreases to a low, approximately constant rate contributed by random failures from less severe design, fabrication, and installation errors. In a narrow sense, these failures can be viewed as aging-related since they were produced by time-dependent degradation. But decades of extensive experience show that such failures occur randomly in time and therefore are not time dependent in terms of failure rate. Reference 3 states that failures in this region "are usually regarded as being independent of degradation due to aging or wearout." Failure rate curves for some components like structures may not be subject to infant mortality and may initially have a small but increasing failure rate until wearout.

Eventually, as denoted by the dashed portion of the curve in Fig. 1, failure rates increase as aging-related failures begin to occur in components that are defect-free. Based on analytical or empirical knowledge of the time it takes aging effects to become significant, the engineer specifies a design life as indicated by the dashed vertical line, at which time the component is refurbished or replaced. This eliminates aging as a cause of failure if there are no unanticipated significant aging mechanisms. If failures occur prior to the design life at a rate greater than the random failure level, corrective maintenance (repair/replacement) is in order and the design life and maintenance interval are reduced to eliminate future age-related failures. Another, perhaps more effective, means of responding to premature failures is replacement with a more rugged component of modified or new design.

Note that the design life indicated in Fig. 1 could refer to components with an original design life on the order of forty years. In this case it will be necessary to project that any extended life period stays within the "design aging margin" shown. On the other hand, for such "long-lived" components, utilities will have to be vigilant to detect any signs of premature aging failures.

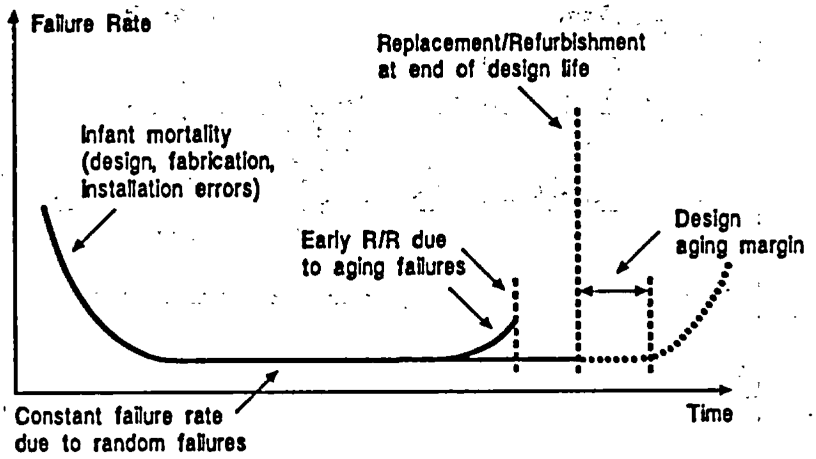


Figure 1. Bathtub-shaped Failure History for Population of Similar Components

A nuclear plant can be viewed as an assemblage of components with a range of lifetimes. Figure 2 is a sketch of failure rate curves for four hypothetical groups of components with various lifetimes. Almost all of the components are shown with a replacement time that precludes aging-related failures. This results from proper design and application, and generally reflects real-world nuclear plant experience. For illustrative purposes, one of the components is shown to experience a rising failure rate due to unanticipated normal aging. Note that subsequent replacements of that component are made at shorter intervals or are of a new design to correct that situation. For components with a longer design life, such discoveries of unanticipated normal aging, if any, will take place later in the life of the plant. Because there is a distribution of lifetimes in plant components, even failures caused by unanticipated normal aging will tend to occur randomly. Therefore, given an integrated plant failure rate that is low enough to provide acceptable overall reliability and safety, there is no rational basis for a plant-wide increase in maintenance for the typical plant situation illustrated in Figure 2.

On the other hand Fig. 2 does illustrate a disadvantage of too frequent maintenance. A smaller refurbishment interval can contribute to greater overall plant failure rate because of more failures from maintenance errors.

The conceptual view that the preponderance of failures can be expected to be random with respect to plant age is confirmed by a recent compilation of U.S. nuclear plant operational experience.[13] Scrams from full power have decreased as a function of plant age for the entire population of plants. This trend is explained by the concepts that (1) failure causes are dominated by random causes from errors and deficiencies which decrease as experience increases and (2) even failures caused by unanticipated normal aging tend to occur randomly. The view that failures are expected to increase with plant age is not supported by the historical trend of scrams.

Role of Aging in Plant Life Assessment

Current understanding of aging and current practices for managing aging are adequate for safe, reliable operation of nuclear plants and support the current 40-year operating license. Since the NPAR program is structured in part to respond to the need "for identifying and resolving technical safety issues related to plant aging and license renewal,"[1] we will address from a utility perspective the role that aging plays in plant life assessment. This perspective is essentially that being used to formulate the extensive U.S. utility efforts in the area of Nuclear Plant Life Extension.[14]

The authors of this paper agree with the NRC that "(aging) is important in the evaluation for license renewal."[1] But this statement must be qualified in that the scope of aging issues important for license renewal is much more limited than implied by this statement and implied in the many NPAR reports to date.

The appropriate focus of life extension efforts should be on

Normal aging of all components with original design lives on the order of forty years or more.

Aging of all components normally replaced during the licensed term (including safety-related ones) has no effect on plant life assessment. These components will simply continue to be replaced regularly during the extended licensed term.

There is no special need to increase attention to abnormal aging in a life extension program. Vigilance for detecting causes (stressors) or symptoms (degradation) of unanticipated aging, whether normal or abnormal, is important regardless of whether license renewal is sought.

Aging management also plays a role in an extended plant term, but that role is no more important than it is during the current term. As mentioned previously,

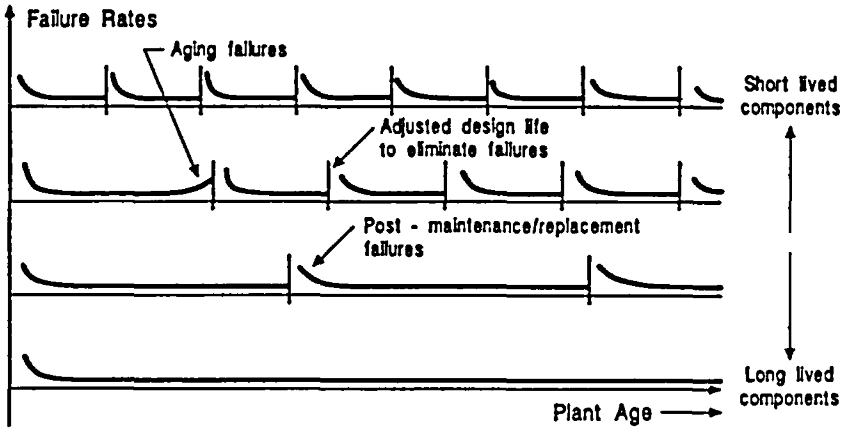


Figure 2. Overall Plant Failure Rate is the Integration of Random Failure of Components with a Wide Range of Design Lives

aging management in an operating plant consists of performance evaluation (ISM and failure trending) and maintenance (preventive and corrective). These address both normal and abnormal aging. The establishment of a level of aging management that is acceptable in terms of safety and economics is an ongoing process as technology matures and operating experience accumulates. There is no rational basis for requiring a wholesale increase in aging management to justify life extension. Specific improvements in ISM of components normally replaced during the licensed term are warranted only where demonstrated to be cost-effective in terms of plant availability. Enhancements of ISM for long-lived components may prove to be desirable, depending on the outcome of residual life assessments. Life assessment research should concentrate on these so-called "major"[15] or "critical"[14] components. Utilities should not lose sight of the fact that resources expended to enhance aging management programs can have a substantial financial payback in the short as well as the long term.

Key aging questions to be posed in a utility life extension program are:

- What components are "critical" (i.e. both safety-related and non-safety-related that are expensive to replace and normally not replaced during the licensed term)?
- What are appropriate criteria for the evaluation of the residual life of these components?
- Are these components monitored and/or included in the maintenance program?
- How long can the original design life of a component be extended when evidence shows that operational stressors were less than assumed for design?
- Can aging degradation be cost-effectively reduced by operational or environmental controls?
- Are there aging mechanisms for critical components that produce significant degradation only after forty years?
- Has there been any abnormal aging that can reduce the additional useful life of critical components?
- Are there technically sound, practical, cost-effective ISM practices in place that can detect unanticipated significant degradation if it occurs particularly for long-lived components?
- What supporting evidence (data, recordkeeping, analysis, inspection, etc.) is needed?

These focused questions are being addressed in a major nuclear plant life extension program being sponsored by EPRI and DOE under the guidance of the NUMARC NUPLEX Working Group.

Conclusions

The general definition of aging as the gradual degradation in the physical condition of a component, system or structure is too broad to be useful for objective assessment of the nuclear power industry's current understanding and management of aging effects. In fact, unfocused use of "aging" as a catch-all issue can be counterproductive to the future of nuclear power in that it (1) implies without technical basis that our understanding and management of aging in operating plants may be deficient, (2) could divert aging research resources away from the most important aspects of aging that warrant attention if we are to extend the licensed term of our plants safely and efficiently, and (3) could divert maintenance resources if overly prescriptive requirements lead to unnecessary or ineffective new monitoring methods or to more frequent than optimum maintenance intervals; the latter could actually increase failure rates by opening the door to more maintenance errors.

A careful examination of component failures in operating plant experience would show that the basic or root cause of the overwhelming majority of failures due to degradation of physical condition relates to the human element, be it in design, fabrication, installation, operation, testing or maintenance. Such causes are defined here and in at least one NPAR report as "abnormal aging." Abnormal aging failures can be expected to occur most frequently during the initial years of plant operation and reduce in frequency as construction errors are uncovered and plant procedures are perfected with experience. Abnormal aging is best reduced by prevention of errors through the continuance or improvement of good operational practices such as vigilance, responsiveness to operational experience, accountability, communication and sound quality programs.

Thus there is no a priori reason why there will be more failures nor why there should be more maintenance as plants advance in age even beyond their current licensed terms. This is supported by plant performance indicators of U.S. plants to date. Neither the need, nor even the technical validity, of constructing mathematical aging terms in probabilistic risk assessments has been established.

The objective of research should be to increase understanding of normal aging of components. The aging of components with shorter design lives is already managed adequately by virtue of its design, qualification, inspection, surveillance, monitoring, refurbishment and replacement. Research should address whether there are aging mechanisms that produce significant degradation only on the order of forty years of operation or longer. Such degradation mechanisms were not addressed in the original design or were assumed to produce insignificant aging degradation during the licensed term. Note that such a focus on unanticipated normal aging would also serve to uncover any aging mechanisms that could conceivably lead to premature failure of long-lived components. Furthermore, it would identify technically sound, practical techniques for monitoring the condition of these critical components. The utility resources needed to implement such techniques would likely have a big payback in short-term and long-term reliability.

Aging should not be viewed as an issue in itself. Such a view is too diffuse and distracts us from focusing on key issues which are:

- Maintenance

  Are there technically sound, cost-effective activities beyond those in place to check for significant unanticipated aging degradation (both normal and abnormal)?

- Life Assessment

  Are there unanticipated aging mechanisms that can impact safe, reliable operation of plants near the end of or beyond current licensed terms?

- Equipment Qualification

  Aging during licensed terms is already addressed by detailed standards and regulations; plant life assessments need examine only the qualified lives of long-lived components such as cable.

In conclusion, we recommend the following:

1. Develop refined, consensus definitions of aging-related terminology. This will improve the accuracy and usefulness of future evaluation and recording of failures, and facilitate dialog about the real issues.
2. Focus future NPAR and industry research on increasing understanding of unanticipated normal aging and opportunities for cost-effective ways to enhance, if necessary, the aging management of critical components.
3. Heighten the current utility/NRC dialog toward the goal of formulating agreed-upon technical conclusions and mutually acceptable recommendations for utility initiatives in response to those conclusions.

References

[1] "Nuclear Plant Aging Research (NPAR) Program Plan," U.S. Nuclear Regulatory Commission, Office of Regulatory Research, NUREG-1144, Revision 1, September 1987.

[2] IEEE Standard 323-74 and -83, IEEE Standard for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations, Institute of Electrical and Electronic Engineers, 1974 and 1983.

[3] S.P. Carfagno and R.J. Gibson, "A Review of Equipment Aging Theory and Technology," Final Report EPRI-NP-1558, Electric Power Research Institute, September 1980.

[4] "Utility Comments on NRC NPAR Program Reports," compiled by EPRI for the Equipment Qualification Advisory Group and NUPLEX Committee, February 1987 and April 1988.

[5] M.S. Kalsi, C.L. Horst, and J.K. Wang, "Prediction of Check Valve Performance and Degradation in Nuclear Power Plant Systems," NRC NPAR Report, NUREG/CR-5159, May 1988.

[6] R.R. Hoopingarner, J.W. Vause, D.A. Dingee and J.F. Nesbitt, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating and Expert Experience," NRC NPAR Report, NUREG/CR-4590, Vol. 1, August 1987.

[7] J. Gleason, "Seismic Ruggedness of Aged Electrical Components," EPRI Final Report NP-501 4, Electric Power Research Institute, January 1987.

[8] B.M. Meale and D.G. Satterwhite, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," NRC NPAR Report, NUREG/CR-4747, Vol. 1, July 1987.

[9] L.L. Bonzon, F.J. Wyant, L.D. Bustard, and K.T. Gillen, "Status Report on Equipment Qualification Issues Research and Resolution," NUREG-CR-4301, SAND85-1309, November 1986.

[10] L.D. Bustard, "Definition of Data Base, Code and Technologies for Cable Life Extension," Sandia National Laboratories Report SAND86-1987, March 1987.

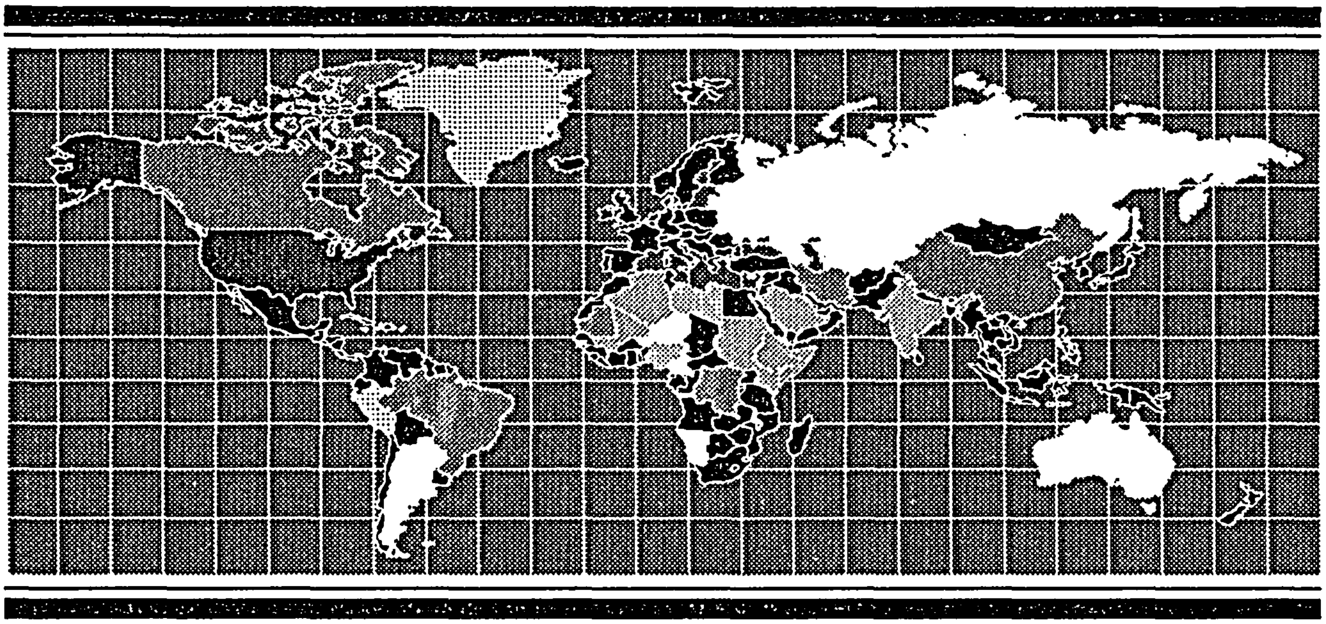
[11] G. Sliter, "EPRI Cable Aging Research," NRC International Nuclear Power Plant Aging Symposium, Bethesda, MD, August-September 1988.

[12] W.E. Vesely, "Risk Evaluations of Aging Research Phenomena: The Linear Aging Reliability Model and its Extensions," NRC NPAR Report, NUREG/CR-4769, April 1987.

[13] R. Koppe, "Use of Performance-Based Safety Evaluation as a Tool for Life Extension," ANS Topical Meeting on Nuclear Power Plant Life Extension, Snowbird, Utah, July-August 1988.

[14] "LWR Plant Life Extension," EPRI Final Report NP-5002, Electric Power Research Institute, January 1987.

[15] V.N. Shah and P.E. MacDonald, "Residual Life Assessment of Major Light Water Reactor Components - Overview Volume 1," NRC NPAR Report, NUREG/CR-4731, June 1987.



TECHNICAL SESSION 2

Aging of Structures and Mechanical Equipment

August 30, 1988

Session Chairman

PROF. DR. KARL KUSSMAUL
Director, Staatliche Materialpruefungsanstalt an der
Universitaet Stuttgart
Federal Republic of Germany

Session Co-Chairman

ROBERT BOSNAK
Deputy Director, Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

K. Kussmaul, J. Föhl
University of Stuttgart, F.R.G.

HOW THE FEDERAL REPUBLIC OF GERMANY IS ADDRESSING THE ISSUE OF AGING OF LIGHT WATER REACTOR COOLANT SYSTEMS

1. Introduction

To ensure intensive cooling, optimum pressure and temperature for both BWRs and PWRs are set. These parameters and the envisaged thermal output determine the design requirements. The design of coolant systems comprises the style of the components, selection of materials, loading parameters resulting from normal operation, as well as upset, emergency, and faulted conditions.

As far as possible, plant aging is being implemented in the design to cope with the potential aging mechanisms. In the areas of mechanical and thermal loading, a reliable stress analysis is needed. In this respect, the pressure-temperature-time histories according to the operation practices determine the usage factor for fatigue which, in many cases, is not well balanced regarding the different stressors in the different components. Corrosion impact may dramatically exhaust both static and cyclic loading capacity. This becomes even more critical when irradiation embrittlement and interaction of the coolant and the stresses occur in the components.

The safety margin during life is dependent on a variety of parameters and may differ through the implementation of different Codes and Standards. The style of the components; selection of adequate materials and procedures for processing, fabrication, and especially welding; as well as quality assurance are decisive prerequisites to assure the required safety margin.

Unavoidable uncertainties can be covered by the assumption of lower bound characteristics. With knowledge of the worst case condition, the minimum required safety margin can be evaluated, and thus the problem of extrapolation to long-range operation eases off.

As a consequence of these considerations, the Basis Safety Concept was developed in the Federal Republic of Germany to render the probabilistic approach unnecessary for safety cases related to catastrophic failure of pressure boundary components (Fig. 1).[1] The process of evaluation started in 1972, and in 1977 the Basis Safety Concept was adopted in principle by the German Reactor Safety Commission (RSK). In 1979, it was officially published and thus became a legal requirement.[2] The essence of this concept is the mechanistic evaluation comprising analysis and testing. The probabilistic evaluation, however, is still important, e.g., in conjunction with the reliability of detection and sizing of flaws through nondestructive examination (NDE).

Apart from quality through design and production, the Basis Safety Concept is based on the existence of the four independent redundancies outlined in Fig. 1 as imperative requirements to rule out the possibility of catastrophic failure and to establish the "incredibility of catastrophic failure principle." The redundancies are:

- Quality assurance through multiple-party testing,
- Plant monitoring,
- R&D work with regard to worst case and failure investigation, and
- Validation of Code requirements as well as fracture mechanics concepts and NDE methods.

Components which do not comply with the Basis Safety requirements must be treated case by case according to the degree of deviation from optimum design and the amount of degradation during manufacture and operation. The redundancies of the Basis Safety Concept can compensate for this lack of compliance, if they are implemented to an effective extent.

2. Aging Phenomena

Pressure-retaining components of a nuclear power plant are subjected to a variety of aging mechanisms which are, with the exception of neutron irradiation, well known from the operation of conventional plants (Fig. 2). Considering components essential for safety, it becomes obvious that fatigue and corrosion are the most frequently acting aging parameters (Fig. 3). The remaining life results from the amount of already absorbed loading capacity. Therefore, the shorter the time to the target life, the higher are the requirements with regard to validated methodologies and accuracy to assess the safety margin and remaining life, respectively, considering time-dependent degradation.

The strategy to cope with aging phenomena includes:

- Evaluation of load-bearing capacity of materials of different quality,
- Improvement and validation of methodologies to understand fracture behavior of specimens and complex structures,
- Improvement and validation of NDE methods,
- Evaluation of relevant aging parameters and quantification of material degradation, and
- Evaluation and implementation of plant-specific measures to mitigate aging phenomena and to realize restoration methods.

3. Material Qualification

The key to the assessment of the minimum required material properties and worst case conditions is the investigation of a wide range of materials in dimensions comparable to original components to include all processing and fabrication parameters which might deteriorate the quality. This was facilitated within the German research program "integrity of components"

(Forschungsvorhaben Komponentensicherheit FKS).[3] The material spectrum covers a range of upper-shelf energy from about 200 J down to about 40 J and of transition temperature (NDT) from -40°C up to 250°C (Fig. 4). The large quantity of material rendered possible extensive testing and statistics. Large-scale specimen testing was realized with CT specimens up to B = 500 mm, crack arrest specimens up to B = 170 mm, notched wide plates W = 200, B = 600, round bars up to 500 mm diameter, and hollow cylinders 800 mm o.d., with wall thickness 200 mm. The materials partially represent rejects which did not meet the requirements for nuclear application and tailored heats up to 200 t aimed in particular at worst case properties related to microstructure, segregation, and crack pattern as well as toughness range. Heavy section welded joints were also produced to cover the similar variety of material states (Fig. 5).

4. Material Characteristics and Component Behavior

For the qualification of a component, material properties are obtained from small specimen testing. For the assessment of the safety margin, those material data can only be applied to complex flawed structures when reliable size- and geometry-independent material characteristics are known.

For the evaluation of failure behavior of specimens and components and the validation of fracture mechanics concepts, a variety of test procedures, test facilities, and measuring equipment has been developed to apply static, dynamic, and cyclic loading in a wide range of temperature, geometry, and size. Besides the outstanding facilities and infra-structure (Fig. 6), special features to be mentioned here are:

- Large-scale specimen test machine (Fig. 7),
- High-speed test machine for large specimens (Fig. 8),
- Pressurized thermal shock (PTS) loop (Fig. 9),
- Full-scale reactor vessel for NDE validation (Fig. 10),
- Dimensions of test components (Fig. 11 examples).

Most of the R&D activities have been carried out in collaboration with research institutes, industry, and utilities with financial support from the Federal Government. The international cooperation and exchange of information and data as well as participation in cooperative programs have assured a sufficiently high level of knowledge and application of advanced methodologies.

4.1 Fracture Mechanics Assessment

A major advancement in fracture mechanics analysis was achieved by introducing the J-integral[4] and adequate procedures to evaluate data for crack initiation and crack resistance of a material.[5] With those parameters it is possible to apply data derived from small-scale specimen testing to large components and to quantify component behavior even beyond crack initiation. This method was validated in the most complex loading situation of PTS where the amount of stable crack extension of the test piece (Fig. 9) was in good agreement with the crack growth derived from a 25-mm-thick side grooved CT specimen.[6] With the available spectrum of materials of different toughness levels, lower bound and worst case conditions are being approached. The evaluation of crack arrest characteristics, however, still suffers from the lack of a validated and generally accepted methodology.[7]

4.2 Leak Before Break

The assurance of leak before break at least for piping is a generic safety issue. Due to the high amount of stored energy, the investigation of failure behavior and failure mode requires great effort. Full-scale straight pipes, elbows, tees, and branches with different flaw configurations and sizes were investigated (Fig. 12).[8] Again, the important parameter is the material toughness in conjunction with the flaw size, for which limited leak before break can be assessed (Fig. 13).

4.3 Plant Tests

Excellent opportunities to perform tests under system and full-scale conditions of primary ferritic and austenitic piping and reactor pressure vessels are provided by the decommissioned HDR plant.[9] Blow-down tests served to evaluate the resistance of piping and core barrel to that kind of loading. The nozzle area of the reactor pressure vessel (Fig. 11) was subjected to severe thermal cycling, whereas for the cylindrical wall, PTS transients were applied.[10] The tests carried out so far represent critical environment conditions due to the high oxygen content of the pressurized hot water at a level of 8 ppm. The continuation of research on that plant (Fig. 14) is an important link in the transferability chain bridging the gap from laboratory test to operating plants with aged components.

4.4 NDE Validation

Although materials in accordance with the Basis Safety Concept provide the necessary level of safety even for flawed structures, it is essential to demonstrate the suitability of nondestructive flaw detection methods together with data acquisition and data processing for sizing, imaging, and interpretation.[11,12] Particularly with regard to older plants which are not designed according to the Basis Safety Concept, enhanced NDE has to compensate for this lack of compliance.[13] However, even in the case of optimized materials, long-range corrosion-assisted cracking also has to be considered for the pressure vessel if through-clad cracks cannot be excluded. The full-scale vessel (see Fig. 10) serves as NDE validation and training center. The vessel includes a broad spectrum of natural and artificial defects in the base material and in the weldment.[14] Areas with complex geometry and difficult access are represented by different design of the nozzles and the penetration on the lower head. At present, the investigation focused on ultrasonic inspection. Pressurizing the vessel, acoustic emissions equipment can be qualified additionally.

5. Time-Dependent Material Degradation

As outlined in Fig. 3, critical conditions may occur when corrosion and/or neutron irradiation become relevant.

5.1 Neutron Irradiation

The potential for material degradation from neutron irradiation was recognized at a very early stage, and world-wide activities have provided the necessary information to cope with this problem. Plant-specific surveillance programs supplement the general knowledge, so that with a sufficient lead factor, the state of the reactor pressure vessel can be evaluated. Although the Heavy Section Steel Technology Program (HSST) in the United States and the FKS program in the Federal Republic of Germany focus on irradiation phenomena, several issues remain unsolved which have to be treated. The joint United States Nuclear Regulatory Commission (USNRC) - MPA-Stuttgart investigation of trepans removed from a German decommissioned pressure vessel (Fig. 15) is aimed at possible effects of neutron energy spectrum, neutron flux, and other parameters which might be different between surveillance specimens and the vessel wall.[15] The relevance of individual chemical elements with regard to the sensitivity of the material and the accumulated neutron dose are being investigated in more detail, implementing microstructural phenomena to either confirm the correlation in use or to evaluate improved Trend Curves, e.g., NRC Regulatory Guide 1.99 and the German KTA 3203 (Kerntechnischer Ausschuss). Materials fabricated in agreement with the requirements of the Basis Safety Concept have shown very small sensitivity to degrading mechanisms during processing and fabrication, and are insusceptible to neutro irradiation as well.

5.2 Corrosion

Life-limiting factors arising from corrosive hot water environment have been well established from the operation of conventional plants.[16] Therefore, corrosion-resistant materials were selected for regions exposed to stress corrosion and corrosion-erosion attack, e.g., stabilized austenitic stainless steel cladding for the RPV and for large-diameter piping in the primary circuit, stabilized austenitic pipes of smaller diameter, and modified Incoloy 800 for steam generator tubes.[17] However, in the case of the RPV and cladded piping, for safety analyses through-clad cracks cannot be excluded, and therefore the behavior of the ferritic material in a hot water environment is of decisive importance not only for the RPV and cladded piping, but also for the other uncladded ferritic components exposed to high-oxygen hot water. After the incubation period, cracks will initiate and grow even under static, monotonic, or slowly alternating cyclic load.

Studies in autoclaves (Fig. 6) under different loading conditions have shown the relevance of stress intensity, loading history, water chemistry, chemical composition of the material (mainly sulfur content), and temperature with regard to crack initiation and crack growth.[18] In this area, the investigation of cracked components (mainly pipe sections) removed from nuclear power stations was a valuable contribution to the understanding of corrosion phenomena and the prevailing parameters.[19] Stagnant hot water was found to be a severe environmental condition. Extensive research programs have been initiated[18,20] to understand corrosion impact in more detail, especially on ferritic steel (Fig. 16). It became obvious that optimized materials (with significantly reduced macro and micro segregations) have a beneficial effect also with regard to corrosion. Although the understanding of corrosion mechanisms has improved in the past, corrosion is still a governing aging process to limit the life of a component and needs further investigation. In the case of corrosion/erosion effects which was also a generic issue in the past, adequate water chemistry, reduction of flow rate, and ferritic material with increased chromium content led to the solution of this problem.

6. Measures To Cope with Aging Phenomena in Operating Plants

The efforts of the German nuclear community focused on aging problems at an early stage (Fig. 17).[21] While at the beginning of the nuclear era activities were concentrated strongly on irradiation effects, today corrosion constitutes the prevailing concern.

Consolidated knowledge was applied to plants under construction, operating plants, and the design of new plants. The measures taken can be subdivided into different areas, namely:

- Rejection of semifinished products,
- Optimization of design and quality assurance as outlined in the Basis Safety Concept,
- Enhanced in-service inspection and monitoring,
- Mitigation methods,
- Restoration methods (repair),
- Replacement of components and systems.

To maintain and assure the high safety standard in nuclear power generation which has been developed in Germany, plants were upgraded according to the state of the art of science and technology.[13,17] The actions taken not only caused high costs, but also required special research and development. Components and systems were replaced, such as steam generators, piping, feedwater vessels, shut down vessels, and core shroud bolts, to mention only a few. Mitigation measures were introduced early to assure safety throughout the design life time. Some of the most important actions are listed in Fig. 18.

7. Conclusion

The safety margin of the pressure-retaining boundary of a nuclear power plant is designed to be at a certain level at the beginning of life (BOL). It must include worst case considerations with regard to material and flaw state as well as loading conditions, and it must also include a reserve to cope with time-dependent material degradation.

In the Federal Republic of Germany, the nuclear community was aware of the safety-relevant parameters, at least for an intermediate time period, and of the fact that supplementary research has to be done to assure sufficient safety for long range-operation.

The implementation of the Basis Safety Concept and the additional independent redundant measures have provided a high BOL safety margin which could be verified through the extensive R&D work. Moreover, the sensitivity to aging parameters could be reduced, in particular with regard to irradiation, and to some extent also to corrosion. Along with the development of the state of the art in science and technology, some components of older plants no longer complied with the high standards of today set by the German overall safety strategy (non-physical aging).[22] Upgrading measures such as replacement and mitigation of operating conditions, however, have increased the safety margin to such an extent that lifetime extension even far beyond the design life becomes feasible.

References

1. K. Kussmaul
   German Basis Safety Concept rules out possibility of catastrophic failure, Nuclear Engineering International, Dec. 1984, pp. 41-46

2. RSK Guidelines for Pressurized Water Reactors, 3rd. Edition, October 1981 (edited by Gesellschaft für Reaktorsicherheit (GRS) mbH, Köln, Germany F.R.)

3. K. Kussmaul
   Aufgaben, Ziele und erste Ergebnisse des Forschungsprogrammes Komponentensicherheit, VGB Kraftwerkstechnik 60 (1980) 6, S. 438-49

4. J.R. Rice
   A path independent integral and the approximate analysis for strain concentration by notches and cracks, Trans. of the ASME Journ. of Appl. Mechanics 35 (1968), pp. 379-86

5. E. Roos and U. Eisele
   Determination of material characteristic values in the elastic-plastic fracture mechanics regime by means of J-integral crack resistance curves, Intern. Journ. of Testing and Evaluation 16 (1988) 1, pp. 1-11

6. K. Kussmaul and A. Sauter
   Widerstandsfähigkeit von Kraftwerkskomponenten 6. Intern. VGB-Konferenz "Forschung in der Kraftwerkstechnik 1988" 16. und 17. März 1988, Essen, Deutschland (FRG)

7. K. Kussmaul and R. Gillot
   Determination of crack arrest toughness at high temperature using compact specimens, Journ. of Pressure Vessel Technology Vol. 110 (1988), pp. 129-136

8. D. Sturm
   Crack behavior of pipes under internal pressure and external bending moment, 4th Japanese-German Joint Seminar, Sept. 21-22 (1988), Kanazawa, Japan

9. K. Kussmaul, E. Roos, H. Diem and G. Katzenmeier
   Validation of pressure boundary structural analysis at the HDR LWR plant, 14th MPA-Seminar Oct. 6-7 (1988), Stuttgart, Germany F.R.

10. R. Stegmeyer, A. Sauter, A. Höfler, J. Sievers and H. Kordisch
    3D-FE-Calculations of a Pressure Vessel Under Thermal Shock Loading, 14th MPA-Seminar, Oct. 6-7 (1988), Stuttgart, Germany F.R.

11. K. Kussmaul, U. Mletzko and D. Sturm
    Automatic in-service inspection of a BWL-FSV activities of the PISC III programme, 6th Intern. Conference on Pressure Vessel Technology, Sept. 12-15 (1988), Beijing, China

12. P. Höller (editor)
    New Procedures in Nondestructive Testing, Springer Verlag (1983)

13. K. Kussmaul
    Specific problems of reactor pressure vessel related to irradiation effects, Radiation Embrittlement and Surveillance of Nuclear Reactor Pressure Vessels: An International Study (L.E. Steele, editor) ASTM STP 819 (1983), pp. 86-99

14. K. Kussmaul and U. Mletzko
    Improvements in quality assurance from the PISC programme (with special emphasis on full scale vessel tests), Intern. Conference on Thermal Reactor Safety, Oct. 2-7 (1988) Avignon, France

15. K. Kussmaul, J. Föhl and T. Weissenberg
    Investigation of materials from decommissioned reactor pressure vessel - a contribution to the understanding of irradiation embrittlement, ASTM 14th Intern. Symposium on "Effect of Radiation on Materials" June 27-29 (1988), Andover, MA, USA

16. K. Kussmaul
    Beobachtungen an Hochleistungs-Kesseltrommeln VGB-Mitteilungen 49 (1969) 2, S. 113-22

17. K. Kussmaul, J. Föhl and E. Roos
    Application of advanced material, design and computation technologies to enhance safety, reliability and availability, IAEA Specialists' Meeting on "Recent Trends in Reactor Pressure Circuit Technology" Nov. 25-28, 1985, Madrid, Spain; Pressure Vessel and Piping 25 (1986), pp. 185-215

18. K. Kussmaul and B. Iskluth
    Environmental assisted crack growth in a low alloy boiler steel in high temperature water containing oxygen, 14th MPA-Seminar, Oct. 6-7 (1988), Stuttgart (FRG) to be published in NED (1989)

19. K. Kussmaul
    Mechanics of materials in corrosion assisted low cycle fatigue crack growth analysis, 2nd Intern. Conference on Low Cycle Fatigue and Elasto-Plastic Behavior of Materials, Sept. 7-10 (1987), Munich, Germany F.R.

20. J. Föhl, Ch. Leitz and D. Anders
    Irradiation experiments in the testing nuclear power plant VAK, Effect of Radiation on Materials (H.R. Brager, J.S. Perrin, editors) ASTM STP 782 (1982), pp. 520-49

21. K. Kussmaul
    Developments in nuclear pressure vessel and circuit technology in the Federal Republic of Germany, Structural Integrity of Light Water Components (L.E. Steele, K.E. Stahlkopf, editors) Appl. Science Publishers LTD (1982)

22. K. Gast
    Aspects of aging from a regulatory point of view. Present thinking in the Federal Republic of Germany, Intern. Nuclear Power Plant Aging Symposium, Aug. 30-31 and Sept. 1 (1988), Bethesda, MD, USA

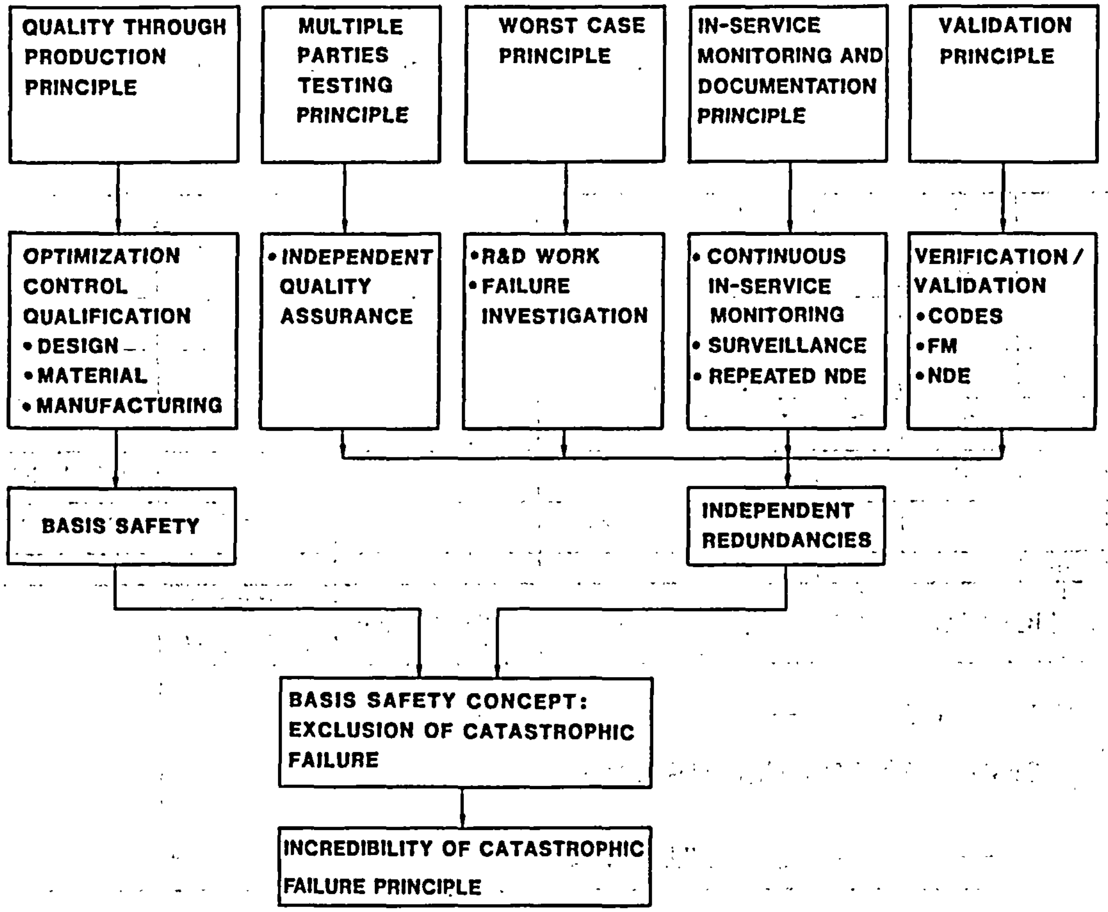


Fig. 1: Summary of the Basis Safety Concept and the "Incredibility of Catastrophic Failure Principle"

	TEMPERATURE								
		NEUTRON FLUX	LOAD: STATIC, CYCLIC, DYNAMIC						RELATIVE MOTION
				LIQUID MEDIUM					
	THERMAL AGEING	IRRADIATION	CREEP		FATIGUE		CORROSION		WEAR
				⏟		⏟		⏟	
REDUCTION OF TOUGHNESS	●	●	●						
GRAIN DISINTEGRATION							●		
CRACKING			●	●	●	●	●		
PITTING							●	●	●
SWELLING		●							
THINNING							●	●	●
DENTING							●		

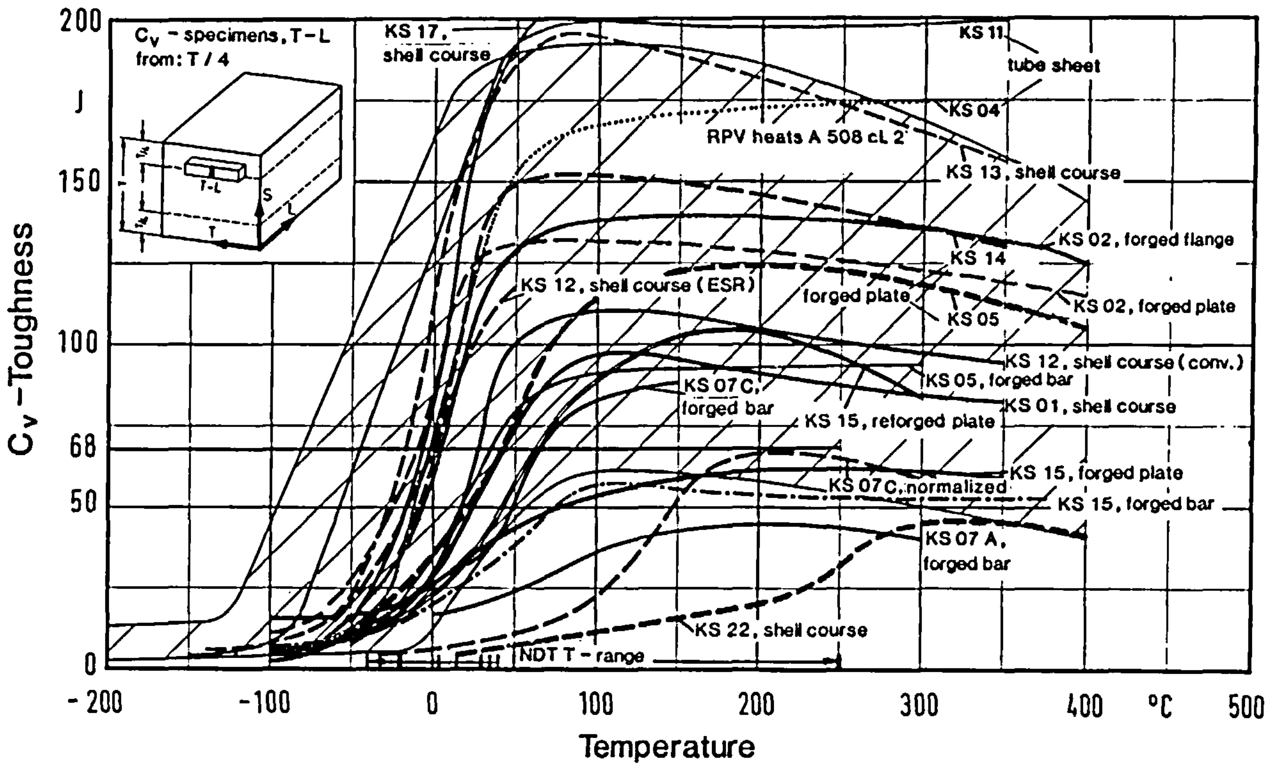
Fig. 2: Parameters relevant for in-service aging potential of materials and components

	TEMPERATURE					
			LOAD: STATIC, CYCLIC, DYNAMIC			RELATIVE MOTION
		NEUTRON FLUX		WATER ENVIRONMENT		
	THERMAL AGEING	IRRADIATION	CREEP	FATIGUE	CORROSION	WEAR
RPV VESSEL SHELL	○	●		○	○	
FUEL ELEMENT CLADDING	○	○	○	○	●	
SG VESSEL SHELL				○	○	○
SG TUBES				○	●	
MAIN COOLANT PIPING				○	○	
MAIN COOLANT PUMP				●	○	
FEED WATER / WET STEAM PIPING				○	●	●
TURBINE				●	○	●
CONDENSATOR TUBES				○	●	

○ POTENTIAL AGEING MECHANISM ● LEADING AGEING MECHANISM

Fig. 3: In-service aging mechanisms of typical light water reactor components

MPA STUTTGART Research Program Integrity of Components (FKS)



<u>Fig. 4</u>: Spectrum of materials investigated in German R&D Programs characterized by Charpy impact and drop-weight testing

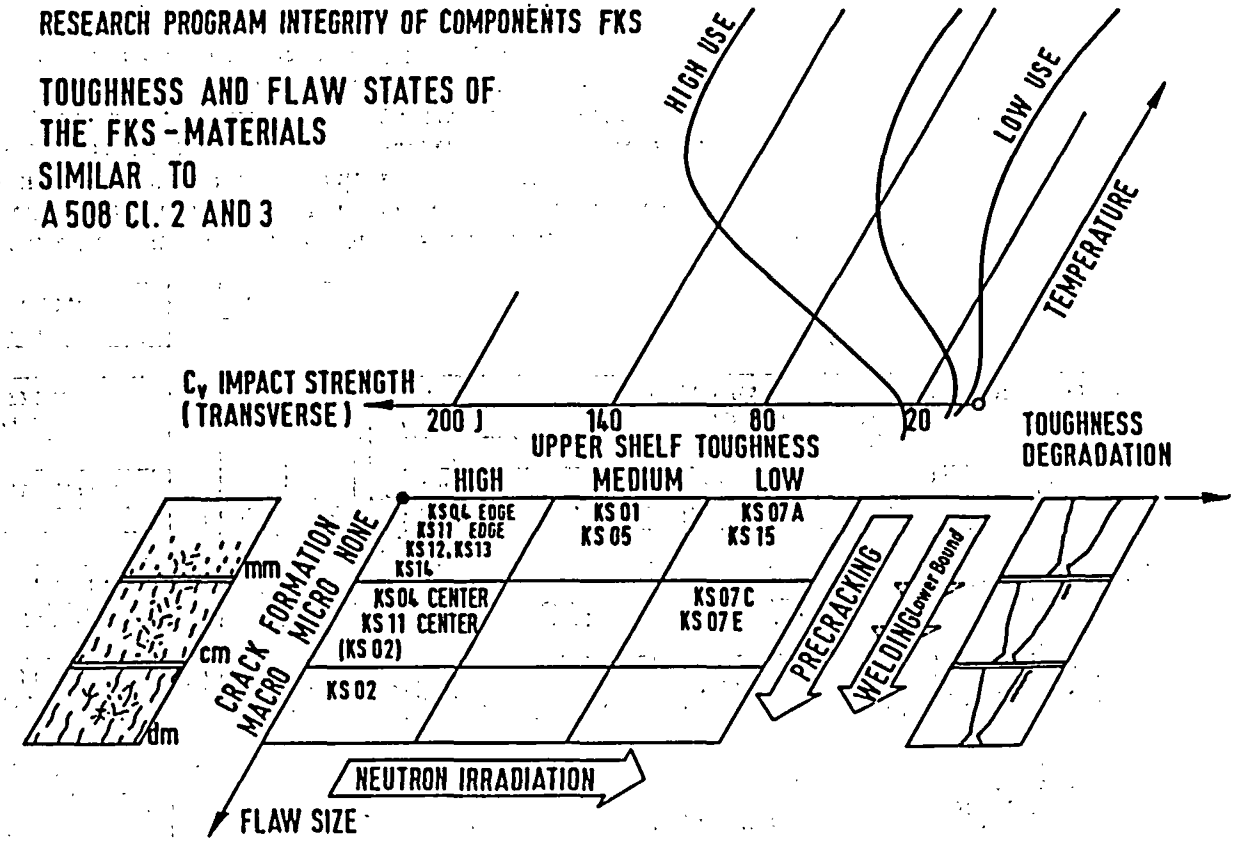
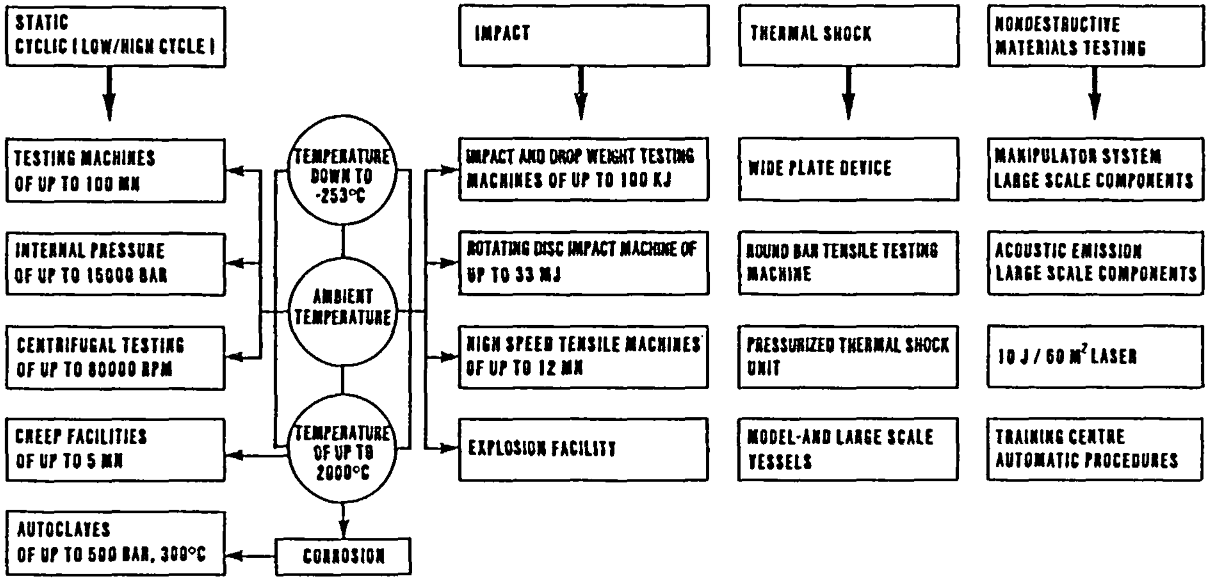


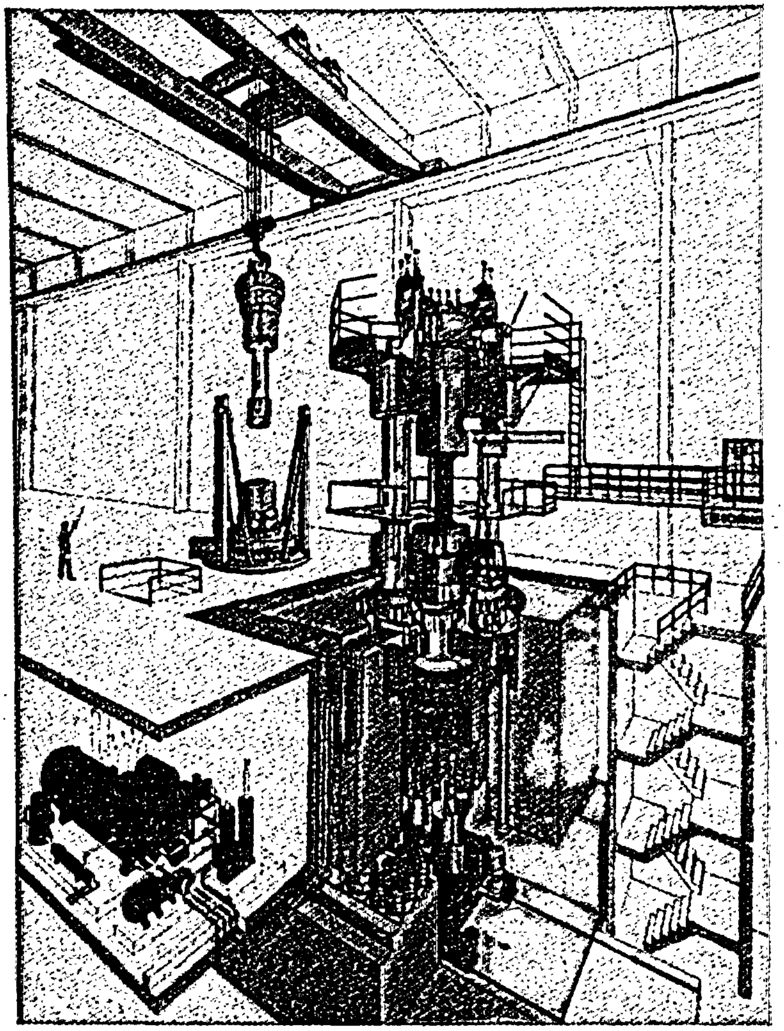
Fig. 5: Matrix of materials covering various toughness and flaw states for base materials and welded joints

SCOPE OF ACTIVITY AND INFRA-STRUCTURE OF MPA STUTTGART



OFFICES, LABORATORIES, TECHNOLOGY TRANSFER CENTRE-AREA COVERED WITH BUILDINGS 11 000 M^2-UTILIZABLE AREA 15 000 M^2
METALLOGRAPHY - ELECTRON MICROSCOPY - MEASUREMENT TECHNIQUES - PHOTOGRAPHIC LABORATORY
TRIBOLOGY LABORATORY - ISOTOPE LABORATORY - WELDING LABORATORY - CERAMICS LABORATORY
ELECTRONIC DATA PROCESSING - SUPERCOMPUTER CRAY 2 - COMPUTER GRAPHICS - IMAGE PROCESSING
MECHANICAL WORKSHOPS AND WELDING FACILITIES FOR MICROPARTS UP TO LARGE SCALE COMPONENTS
OFFICIAL CALIBRATION FACILITY FOR TESTING MACHINES AND FORCE MEASURING INSTRUMENTS
EXTERNAL TESTING AREAS IN MANNHEIM, MEPPEN AND IN THE ENVIRONS OF FRANKFURT

Fig. 6: Test facilities and infra-structure to cope with advanced requirements for material and component qualification

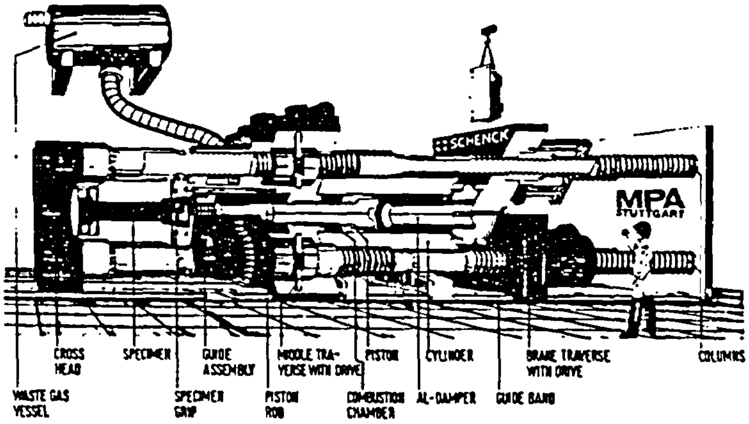


TECHNICAL DATA:

Load capacities	
tension:	*max. 100 MN (22 500 kips)*
compression:	*max. 50 MN (11 250 kips)*
Hydraulic pressure:	*max. 490 bar (49 MPa)*
Piston	
diameter:	*2000 mm (79 inches)*
stroke:	*350 mm (13,8 inches)*
speed:	*max. 80 mm/min (3,2 in/min)*
Hydraulic oil volume:	*approx. 6000 l (1580 gal)*

Electrical supply:	*150 kW*
Total height of the facility:	*approx. 17 700 mm (58 ft)*
Weights	
Round or flat tensile specimen ready for testing:	*approx. 70 tonne*
Machine with specimen:	*approx. 650 tonne*
Reinforced concrete ballast block:	*approx. 600 tonne*
Reinforced concrete sub-foundation:	*approx. 600 tonne*

Fig. 7: 100 MN tensile machine



LOAD	12 MN	**WEIGHT**	
STROKE	600 (1300) mm	TOTAL	490 t
VELOCITY		ACCELERATING PARTS	15 t
-AFTER 20 mm STROKE	25 m/s	ACCELERATION FORCE	100 MN
-AFTER 400 mm STROKE	60 m/s		
LOAD FRAME STIFFNESS	10^{10} N/m		
CROSS-SECTIONAL AREA(Specimen)	20000 mm²	**STRESSES**	
MAX. INTERNAL PRESSURE	200 MPa	PISTON ROD	MAX. 900 MPa
I.D. OF CYLINDER	1200 mm	CYLINDER	MAX. 625 MPa

Fig. 8: 12 MN high speed tensile machine

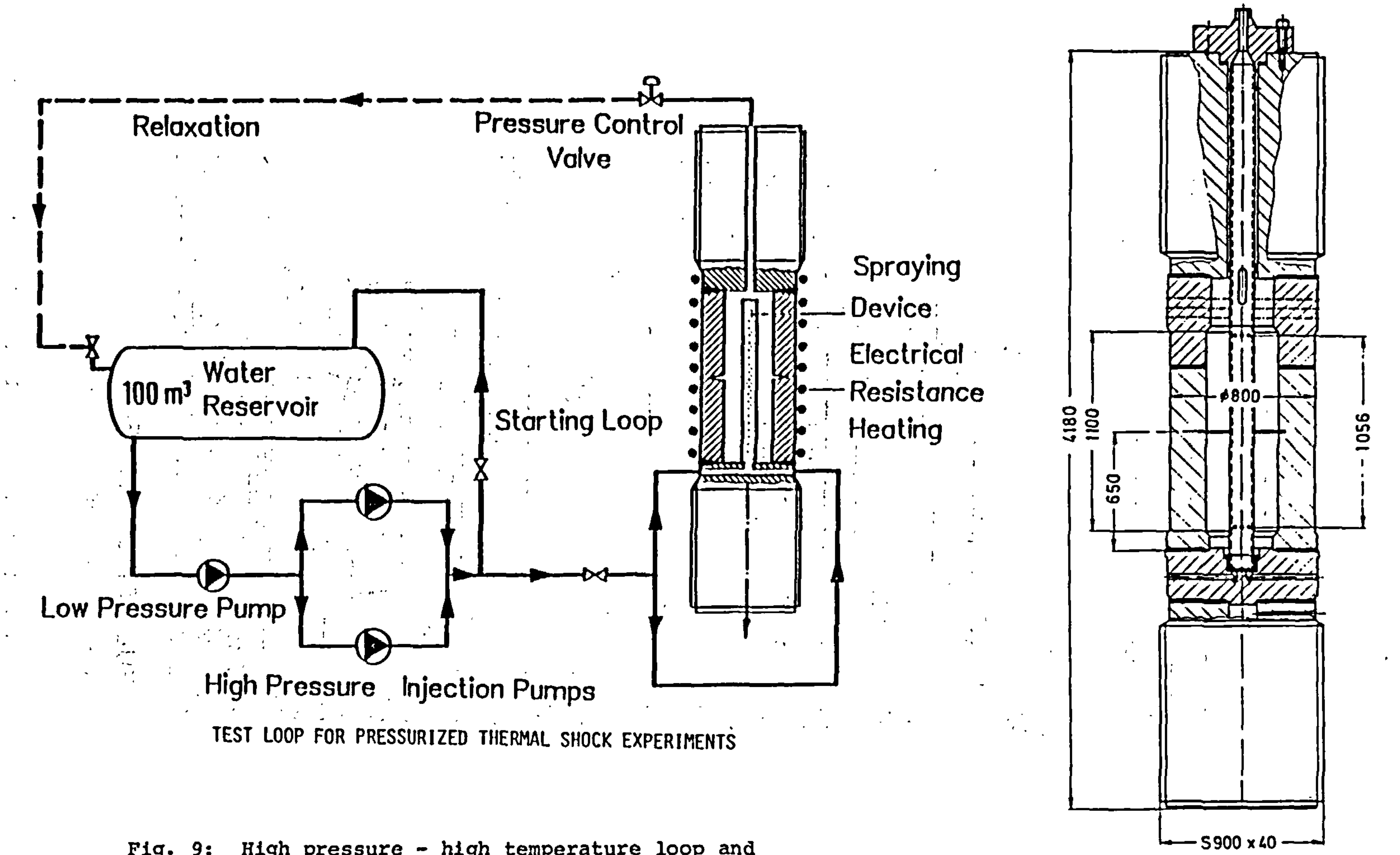
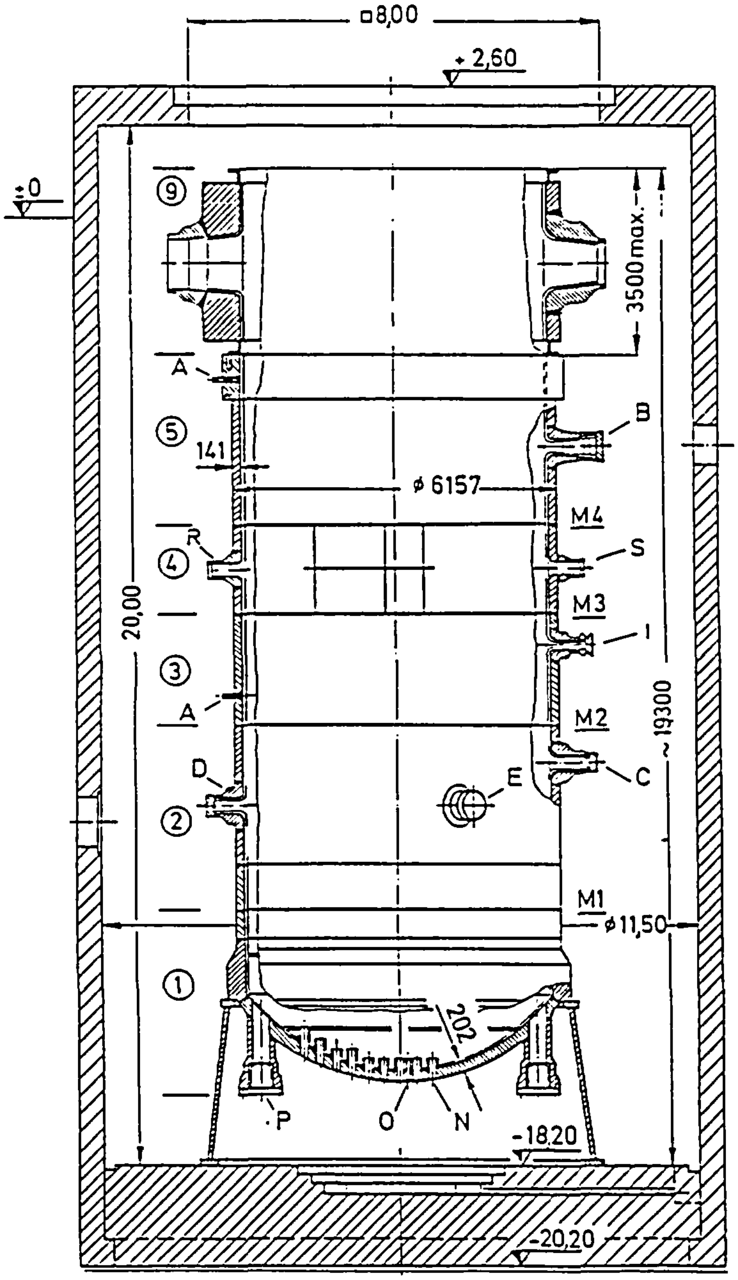


Fig. 9: High pressure - high temperature loop and test specimen (hollow cylinder) to simulate pressurized thermal shock (PTS) conditions



FULL SIZE VESSEL

WITH ADDITIONAL SHELL ⑨

Fig. 10: Full size vessel with BWR and PWR module for NDE validation, e.g. PISC III

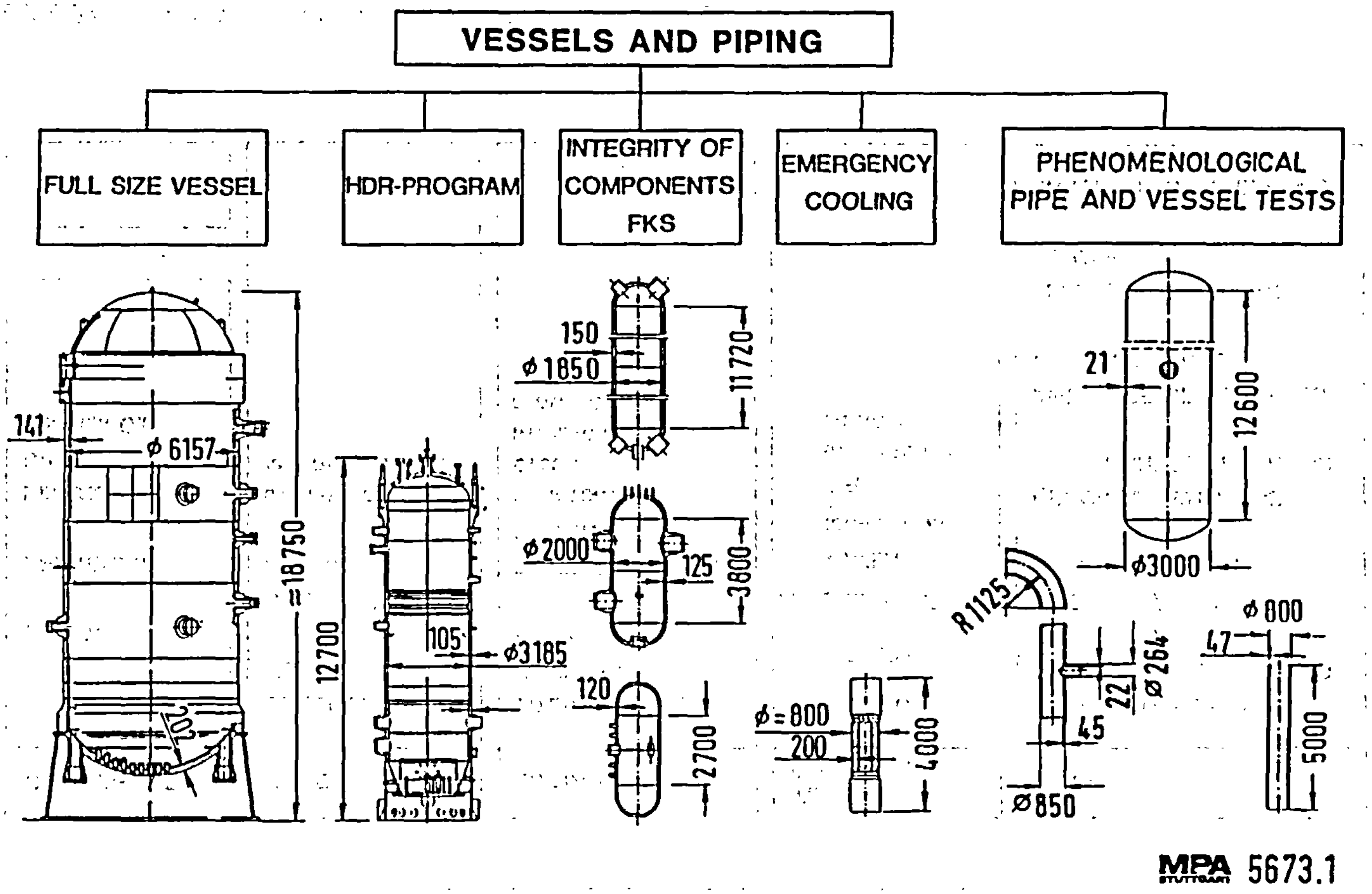


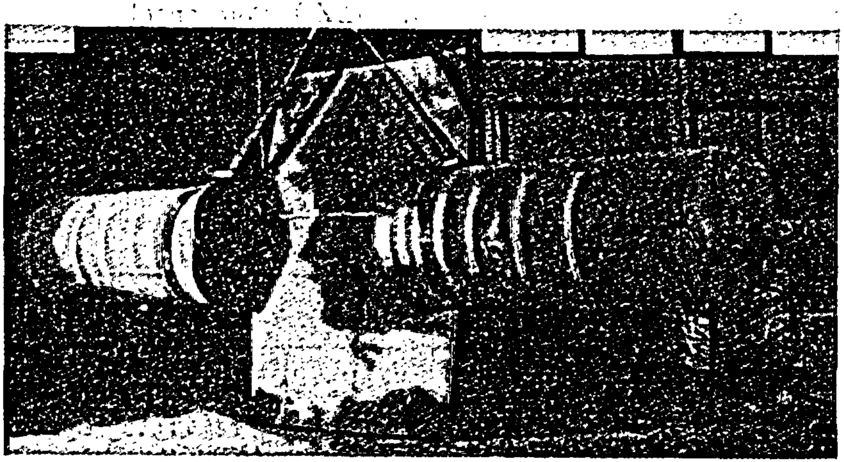
Fig. 11: Vessels and piping used for validation of fracture mechanics and NDE concepts

GERMAN R&D PROGRAM "PIPING"

COMPONENTS	CRACK TYPE	LOADING	TEST CONDITIONS	AIMS
STRAIGHT PIPES ELBOWS BRANCHES DIMENSIONS UP TO 800 mm O.D.	LONGITUDINAL FLAWS CIRCUMFERENTIAL FLAWS (PART-THROUGH AND THROUGH WALL)	INTERNAL PRESSURE MONOTONIC, STATIC EXTERNAL BENDING MOMENT MONOTONIC CYCLIC DYNAMIC (EARTHQUAKE, BLOW-DOWN)	T = 300 °C p_i = 15 MPa PRESSURE MEDIUM WATER (CORROSION) AIR (HIGH STORED ENERGY)	LOAD BEARING CAPACITY OF FLAWED COMPONENTS CYCLIC PLASTIC DEFORMATION BEHAVIOR CRACK GROWTH AND CRACK ARREST BEHAVIOR INFLUENCE OF TOUGHNESS (50 - 150 J USE) LEAK-BEFORE BREAK BEHAVIOR CRACK OPENING BEHAVIOR
SYSTEMS				

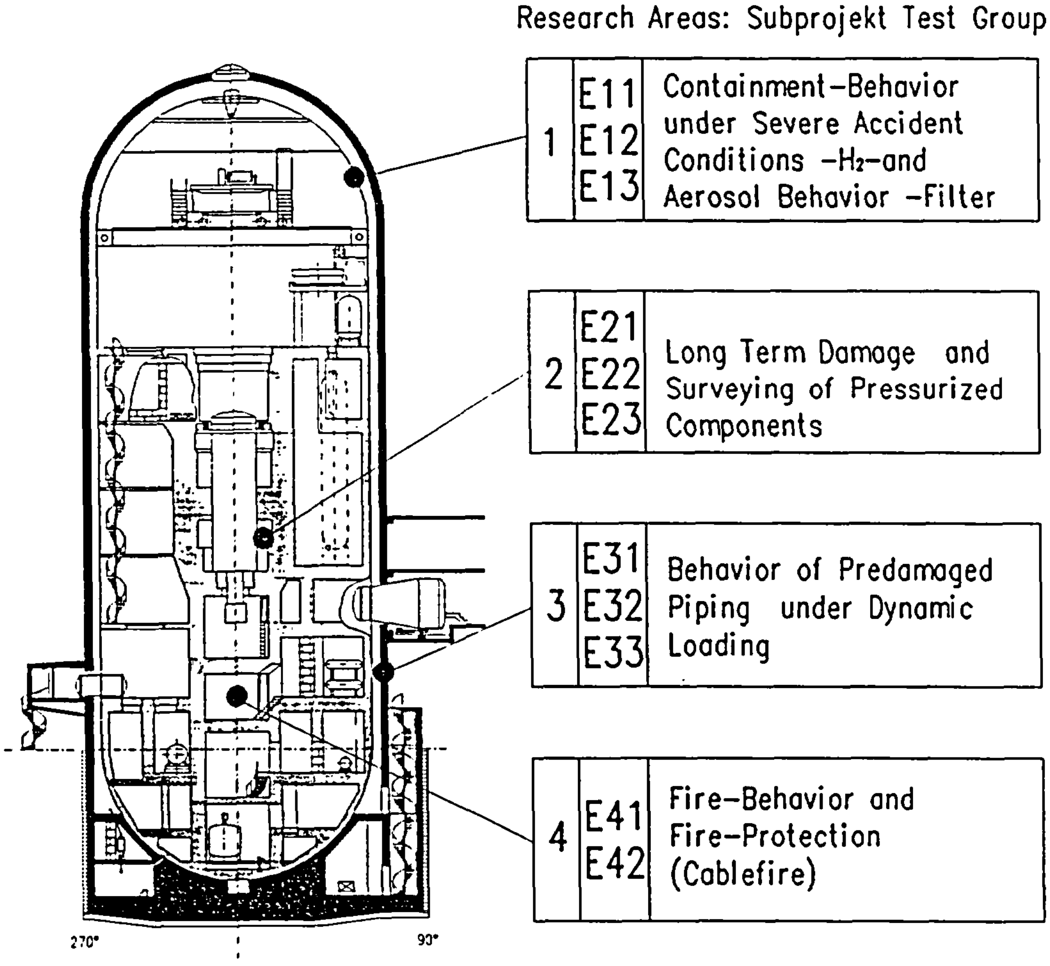
Fig. 12: Main tasks of the German Piping R&D Program including full scale systems (HDR plant)

a)



b)

Fig. 13: Pipes (798 mm o.d., 47,5 mm wall thickness) after pressure test representing limited leak (a) and catastrophic failure (b)



Research Areas: Subprojekt Test Group

1	E11 E12 E13	Containment-Behavior under Severe Accident Conditions $-H_2-$and Aerosol Behavior -Filter
2	E21 E22 E23	Long Term Damage and Surveying of Pressurized Components
3	E31 E32 E33	Behavior of Predamaged Piping under Dynamic Loading
4	E41 E42	Fire-Behavior and Fire-Protection (Cablefire)

acc. to KfK

Fig. 14: Overview of the main research tasks of HDR project phase III

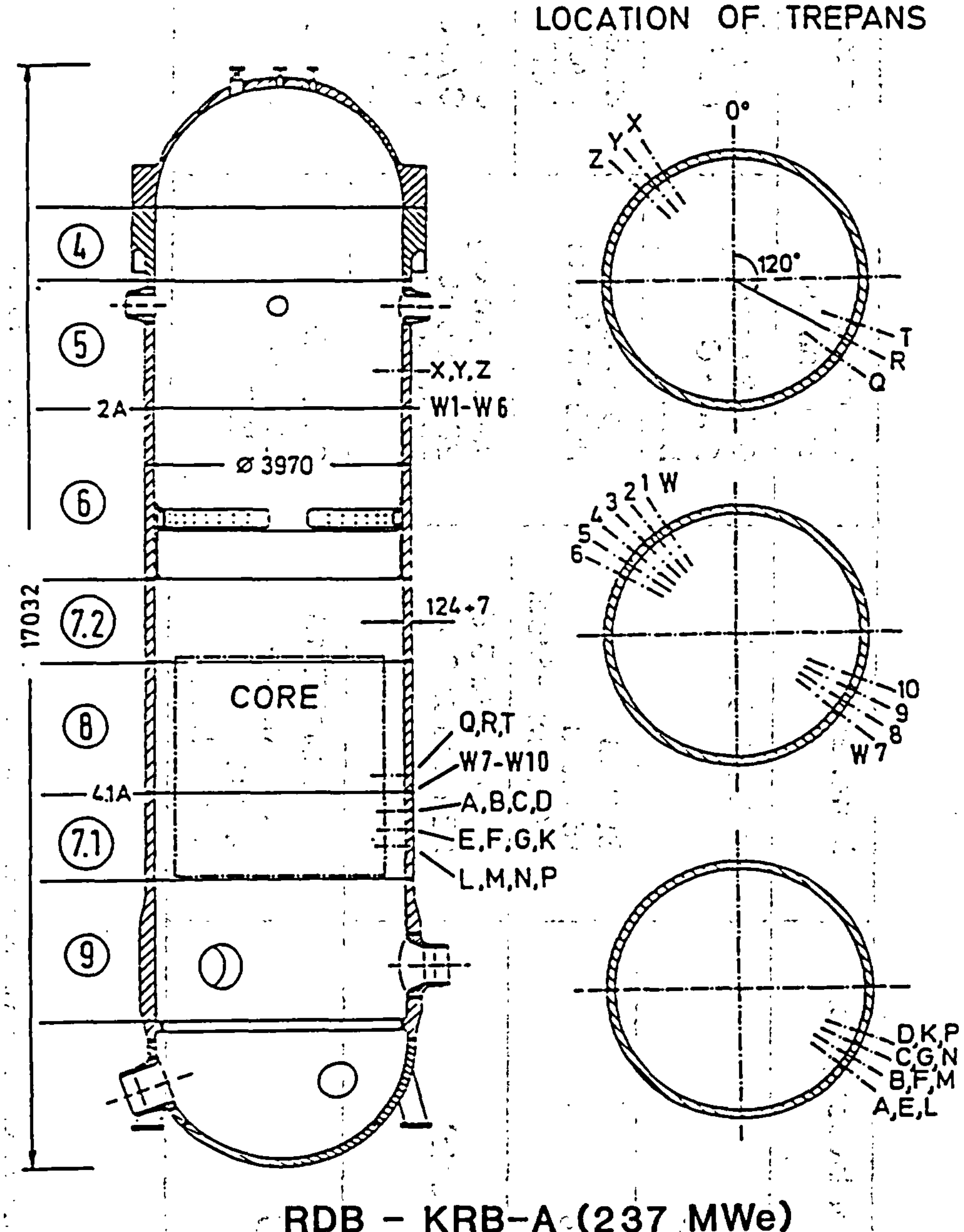


Fig. 15: Investigation of trepans (location of trepans marked with letters for base material, with numbers for weld material) of the decommissioned LWR power plant Gundremmingen unit A (KRB-A) for validation of surveillance practice

Topic of Investigation	Material	Corrosive Environment	Loading
VGB (piping)	MnMoV-Steel	0,4-8 ppm O_2 240 °C BWR	Static LCF
HDR (piping)	St 35,8 III 15 Mo 3 NiCuMoNb-Steel	8 ppm O_2 290 °C Water	Static Cyclic LCF
FKS II (RPV Steels)	20 MnMoNi 5 5 22 NiMoCr 3 7	0,4-8 ppm O_2 200-290 °C BWR, PWR	Static Cyclic LCF CERT
HEW (piping)	X 10 CrNiTi 18 9 A 347 (mod.) A 304 A 316 NG	8 ppm O_2 240 °C BWR	Static LCF

Fig. 16: Main features of the corrosion program of light water reactor components

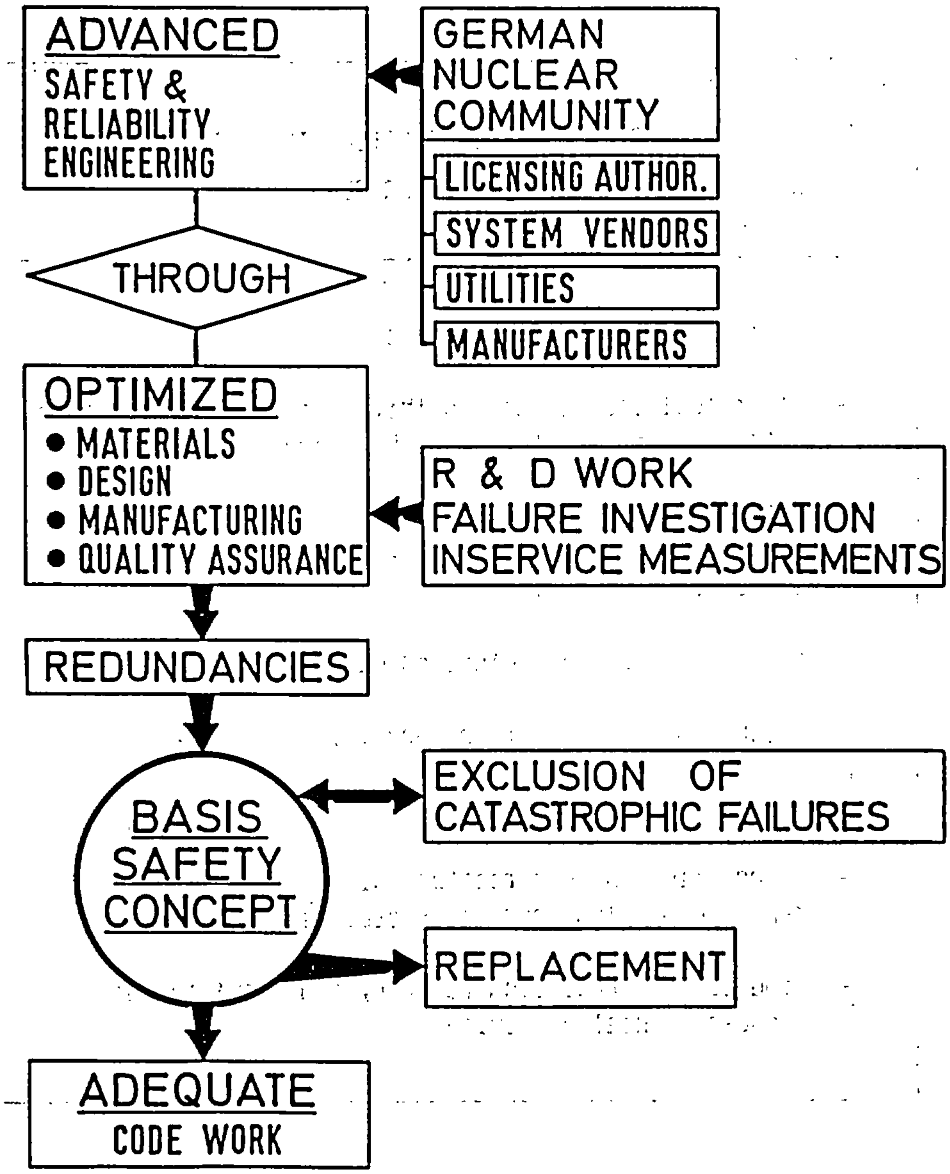


Fig. 17: Main areas of the Basis Safety Concept including upgrading measures of older plants, e.g. by replacement of components

EXAMPLES OF PREVENTIVE MITIGATION METHODS IN NUCLEAR POWER PLANTS

- optimized operating procedure
- electrical locking of prohibited pressure/temperature pathes during start-up and shut-down
- optimized flow conditions of coolant
- improved water chemistry
- additional hot leg injection during ECC
- extension of irradiation surveillance programs
- neutron flux reduction
- irradiation in host reactors with original and reproduced welds (parameter studies)
- studies on irradiation recovery and provision of stand-by annealing device

Fig. 18: Mitigation measures to assure safety of older plants with regard to aging effects

AGING OF STRUCTURES IN AUSTENITIC CHROMIUM-NICKEL-MOLYBDENUM STEELS

Eugeniusz Szpunar, Jerzy Bielanik

1. Introduction

Corrosion resisting and creep resisting steels having an austenitic or austenitic-ferritic structure can change their phase composition during processing, involving welding or plastic working and heat treatment, as well as in the course of their practical service in the temperature range of 600-900°C, this being the result of the precipitation of intermetallic and interstitial phases. The changes in the phase composition can - depending on the kind and number of phases as well as on their morphology and nucleation site - influence essentially the resistance against various kinds of corrosion as well as provoke an embrittlement and a decrease in the mechanical properties.

2. Aim and methodics of investigation

The aim of the present work is to investigate the precipitation processes in austenitic chromium-nickel-molybdenum steels of the 17-14-2 type. According to the present-day state of the knowledge in the field of aging after quenching, precipitation of interstitial phases, particularly of chromium and molybdenum carbides and carbonitrides, is to be expected. With this process, precipitation of intermetallic phases, e.g. σ, χ and η can superimpose. In view of the very great differences in opinions as regards the development of precipitation processes in these steels, identification of the intermetallic phases, precipitating in the course of aging, was carried out; apart from that, an attempt was made to elucidate the role of the high-temperature δ ferrite in the precipitation of intermetallic phases.

The nucleation sites of the phases and their morphology were investigated on thin foils using transmission electron microscopy as well as on microsections using classical light microscopy and scanning electron microscopy. The identification of the precipitated phases was performed by means of electron diffraction analysis on thin foils and on isolated particles on carbon replicas, covered with a standard substance /Au/. The type of the crystal lattice of the phases was determined and the lattice parameters were measured. The establishing of the phase at the cost of which the precipitate grows was accomplished on the basis of coincidence of crystallographic directions and of the degree of lattice fit in the habit plane. Variations of the lattice parameters of austenite as a function of aging temperature and aging time were determined by means of X-ray diffraction analysis.

3. Materials for investigation

The investigations were carried out on the 00H17N14M2 steel of the following chemical composition: 0,028% C, 1,63% Mn, 0,50% Si, 0,018% S, 0,040% P, 17,52% Cr, 12,61% Ni, 2,28% Mo, 0,22% W, 0,21% Cu. As a supplementary material, the 0H17N12M2T steel, stabilized with titanium, was used, its composition being as follows: 0,046% C, 1,68% Mn, 0,46% Si, 0,033% S, 0,002% P, 16,8%, Cr, 11,6% Ni, 2,23% Mo, 0,28% Ti.

The materials used for investigations were quenched from the temperature of 1150°C, this being folowed by aging at temperatures of 600, 700, 800 and 900°C. As the main point of interest in the present work was the nucleation and growth of intermetallic phases, the following aging times were chosen: 2, 12, 50, 150, 450 and 900 hours.

4. Results of investigation

4.1. Structural examination

In as-quenched condition, the 00H17N14M2 steel has a structure composed of twinned austenite and of very minute, spheroidal particles of a precipitated phase which, at this stage of investigation, could not be identified /Figs 1a and 1b/.

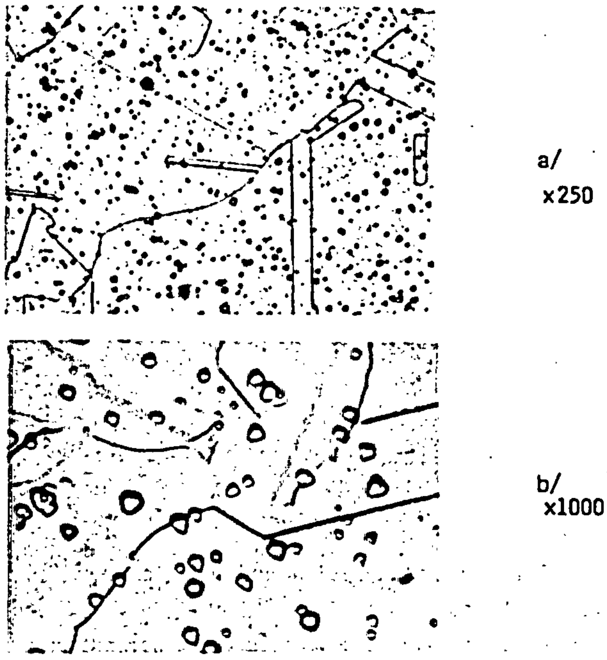


Fig.1. 00H17N14M2 steel in as-quenched condition.

The 0H17N14M2 steel in as-quenched, markedly finer grains /Fig.2a/. Fine lens-shaped and spheroidal particles of the precipitated phase, located at the grain boundaries, suggest this precipitate to be the high-temperature δ ferrite /Fig.2b/.

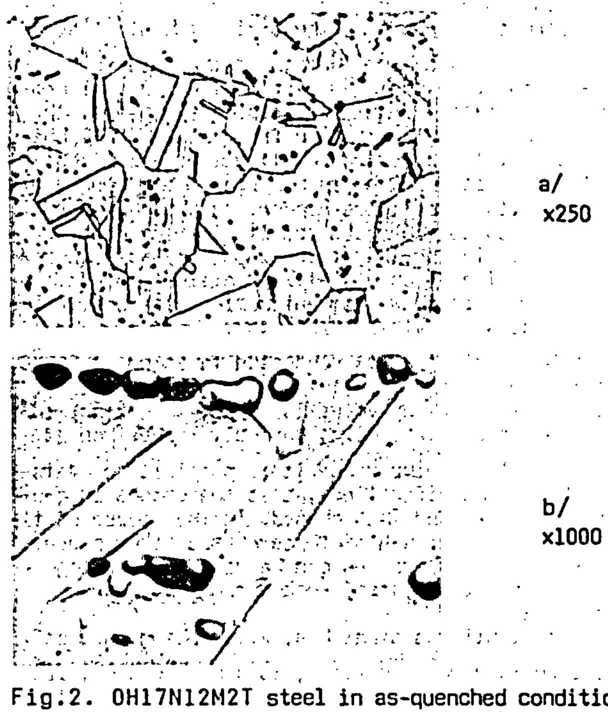
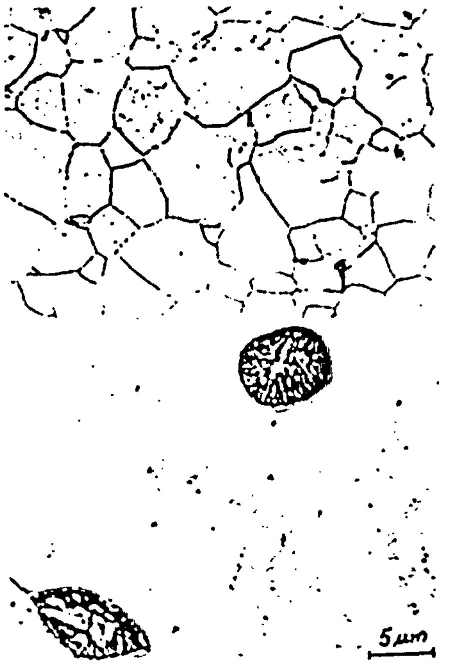


Fig.2. 0H17N12M2T steel in as-quenched condition.

Visible changes in the structure of the steels under investigation were observed after aging at 700°C for 10 hours. In the 00H17N14M2 steel the spheroidal regions within the austenite grains partially vanished while in the remaining spheroidal and lenticular particles at grain boundaries changes were observed pointing to the precipitation of a new phase /Figs 3a and 3b/.



a/
x250

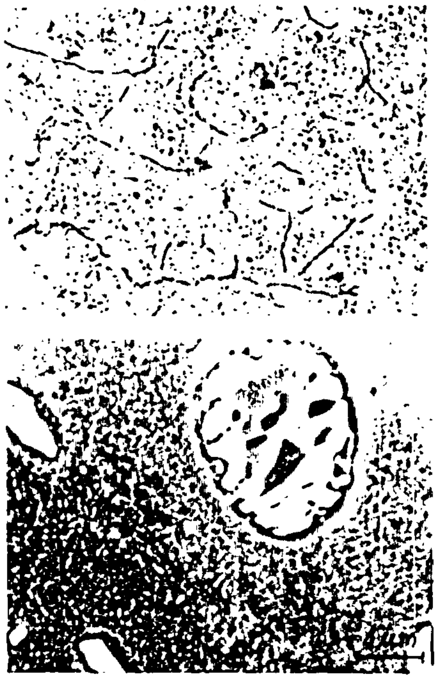
b/
scanning

Fig.3. 00H17N14M2 steel, quenched and aged: temperature 700^{0}C, time 10 hours.

In the 0H17N12M2T steel, at analogous aging parameters, vanishing of twins in austenite was observed while in the regions of supposed δ ferrite changes, similar to those in the non-stabilized steel, were found. At the grain boundaries of austenite numerous fine precipitate particles were present, distinctly observable by means of a scanning electron microscope. An extension of the aging time up to 450 hours resulted only in a slight thickening of the grain boundaries without any noticeable changes in the structure. Aging of the 00H17N14M2 steel at 800^{0}C during up to 450 hours has shown that precipitation process change, to a higher degree, the chemical composition of the matrix and the precipitated phases. An increase in the aging temperature from 700 to 800^{0}C changes essentially the structure of titanium stabilized steels. Already after 150 hours a large number highly dispersed precipitate particles appear. An extension of the aging time up to 450 hours results in obtaining a structure very similar to that found in the non-stabilized steel /Fig.4a/. The structure still contains regions of the supposed δ ferrite within which precipitated particles of new phases are visible /Fig.4a/.

When the 00H17N14M2 steel is subjected to aging at 900^{0}C, precipitated phases appear within austenite as well as at its boundaries already after 10 hours, the dispersion degree of the precipitate being markedly lower then at 800^{0}C. Using a light microscope it is possible to establish, at higher magnifications, that the phases in question are plate-like or lath-shaped. In the course of time, coalescence and growth of these phases takes place. The intensity of this process can be estimated basing on the structure of the steel aged during 900 hours /Figs 5a and 6/.

The titanium-stabilized steel has, after being aged at 900^{0}C, a structure which differs, to a high degree, from that of the 00H17N14M2 steel. At the initial stage of aging /10 hours/ the structure resembles much that in the as-quenched condition. After longer aging times /150 hours/ a specific contrast appears, testifying to the occurrence of minute precipitate particles. After long aging times /900 hours/ the grain boundaries and boundary-adjoining regions become decorated with small spheroidal precipitate particles /Fig.6a/. Regions of the supposed δ ferrite, stretching along the grain boundaries, get strongly etched /Fig.6b/.



a/
x250

b/
scanning

Fig.4. 0H17N12M2T steel, quenched and aged: 800^{0}C, time 450 hours.

a/
x250

b/
x1000

Fig.5. 00H17N14M2 steel, quenched and aged: temperature 900^{0}C, time 900 hours.

The investigation of changes in the lattice parameter of the Feγ solid solution in the 00H17N14M2 and 0H17N12M2T steels, performed using X-ray technique, has shown that these changes are most marked during the first two hours of aging. This refers, especially, to the 00H17N14M2 steel. In the case of the parameter of austenite changes, is extended. The course of the observed changes suggests the conclusion that - during the first hours of the aging process - precipitation of interstitial phases takes place which - after longer times - is followed by precipitation of intermetallic phases.

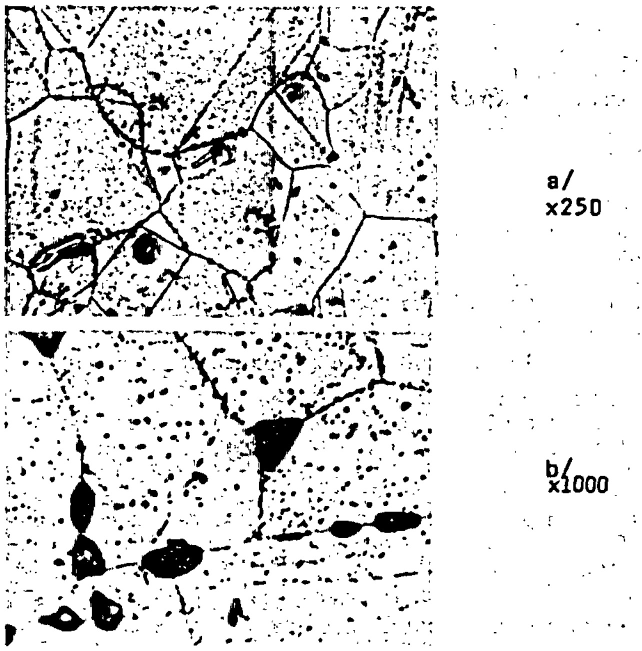


Fig.6. 0H17N12M2T steel, quenched and aged: temperature 900°C, time 900 hours.

4.2. Identification of the phases

In order to get reliable information concerning the observed structures, identification of the phases involved was carried out by electron diffraction techniques using thin foils as well as isolated particles extracted on carbon replicas. In the case of thin foils the investigation embraced the sites of precipitation as well as the distribution and morphology of the phases; apart from that, the phases were identified and their orientation in relation to the solid solution Feγ was established. The examination of thin foils was supplemented by the investigation of isolated particles. These investigations made it possible to determine the morphology of the minute precipitate particles, appearing at the initial stage of aging, as well as to identify them.

In as-quenched condition the structure of the 00H17N14M2 steel was austenitic. Both within the grains and at their boundaries no precipitated particles of interstitial or intermetallic phases were observed / Fig.7/. In the 0H17N12M2T steel, however, in some austenite grains or at their boundaries precipitated phases were found which were identified by electron difraction techniques as being $M_{23}C_6$ carbides /Fig.8a,b and c/. Besides carbides, precipitated particles of the high-temperature δ ferrite occured /Fig.9a,b and c/. In the non-stabilized steel no δ ferrite was observed.

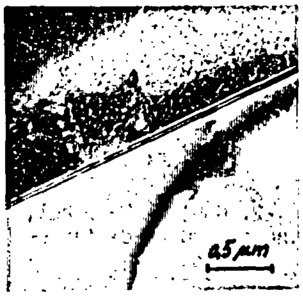
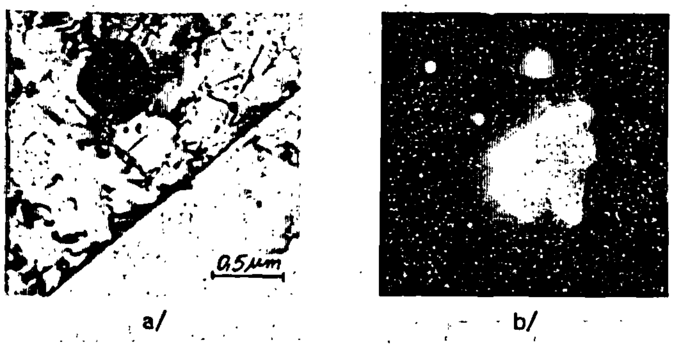


Fig.7. 00H17N14M2 steel in as-quenched condition.

Aging of the 00H17N14M2 steel at 600°C entails changes in the structure, consisting in the precipitation of highly dispersed phases of a plate-like morfology. Diffraction analysis performed on thin foils did not result in the identification of these phases. By using extraction replicas, however, difficulties connected with obtaining diffraction patterns were eliminated. The occurence of a phase with a plate-like morphology was observed, this phase having a cubic lattice with the parameter equal to 6.4 Å.



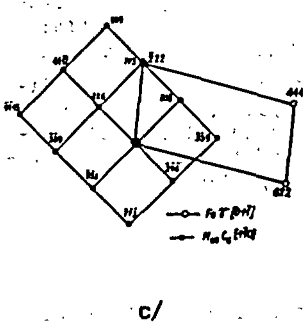
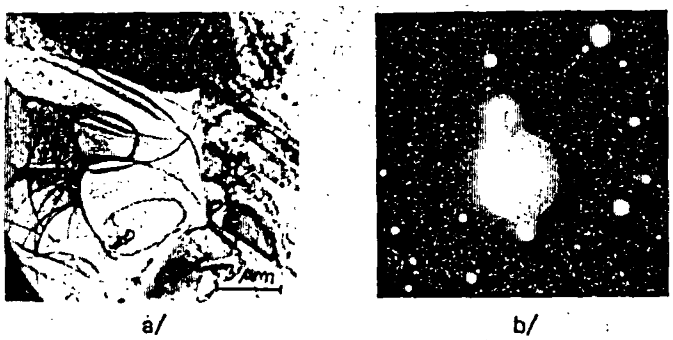


Fig.8. 0H17N12M2T steel in as-quenched condition: a/ not disdissolved $M_{23}C_6$ carbides; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.



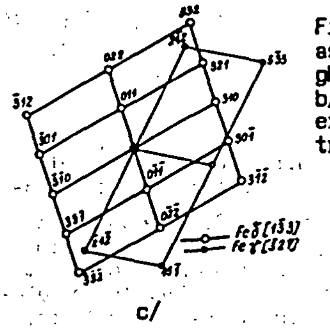
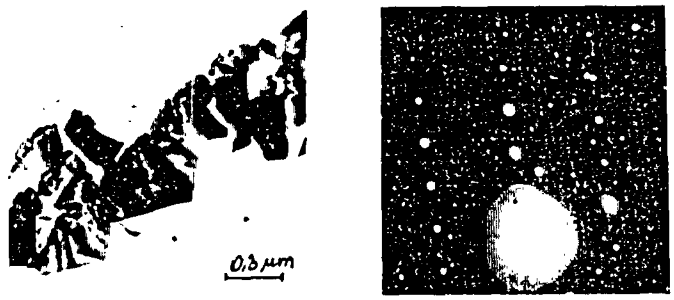


Fig.9. 0H17N12M2T steel in as-quenched condition: a/ high-temperature δ ferrite; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.

Apart from that, agglomerations of particles of another cubic phase were observed, its parameter being 10.62 Å /Fig.10a,b and c/. This phase, with a plate-like morfology, does not have any know equivalent in the literature so far available. The even sum of the indexes indicates that its lattice is body-centered. The agglomerations of particles with a cubic lattice were identified, after solving the diffraction patterns, as the $M_{23}C_6$ carbide.

If aging times of the 00H17N14M2 steel at 600°C are longer, the number of extracted phases distinctly increases. Besides the $M_{23}C_6$ carbide and an unknown cubic phase with the parameter equal to 6.4 Å, other phases appear. On the basis of the type and parameter of the lattice /a=8.79 Å/, the occurrence of the intermetallic χ phase was ascertained.



a/ b/

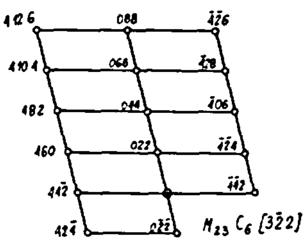
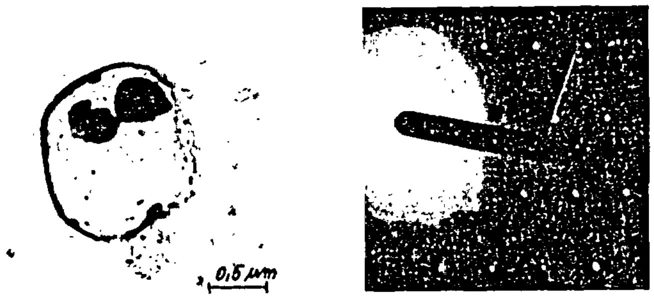
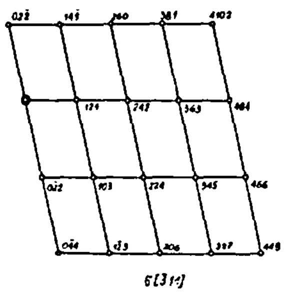


Fig.10. 00H17N14M2 steel, quenched and aged: temperature 600°C, time 2 hours: a/ $M_{23}C_6$ carbide precipitate revealed on an extraction replica; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.

To this phase - in austenitic chromium-nickel-molybdenum steels - the formula $Cr_6Fe_{18}Mo_5$ is ascribed. At the aging parameters in question /600°C, 50 hours/, the most numerous group is constituted by particles situated within characteristic circular regions, impressed in the replica /Fig.11a/.



a/ b/



c/

Fig.11. 00H17N14M2 steel, quenched and aged: temperature 600°C, time 50 hours: a/ σ phase precipitate in the region of the high-temperature δ ferrite; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.

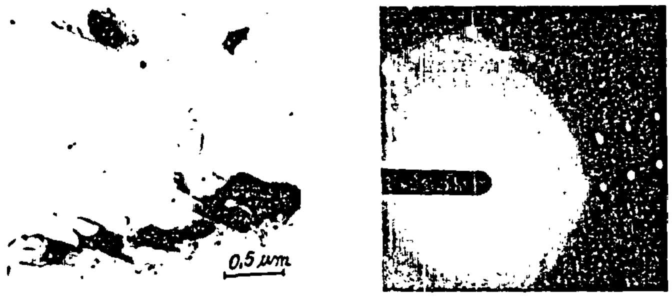
These regions correspond, as regards their shape, to the spheroidal precipitated particles observed in the course of investigation by means of a light microscope /Fig.14/. Numerous diffraction studies have shown that the precipitated particles occurring within the spheroidal regions have a tetragonal lattice, its parameter lying in the range of 8.78-8.81 Å and their tetragonality amounting to 0.52 /Fig.11b and c/. These parameters correspond to the σ phase. The presence of an inconsiderable amount of the $M_{23}C_6$ carbides has also been established.

Structural investigations as well as the identification of phases in the 0H17N12M2T steel, aged at 700°C, showed the evolution of the processes to be similar to that taking place at 600°C, with the difference only that, here, precipitated particles of the M_6C carbide occur occasionally. In the regions identified as the high-temperature δ ferrite the σ phase precipitates.

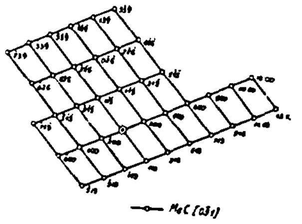
At 800°C the precipitation process is markedly accelerated. In the 00H17N14M2 steel the presence of a plate-like precipitate within the δ ferrite can be observed already after two hours. By means of diffraction analysis it was possible to identify this precipitate as the χ phase with a cubic lattice, its parameter being aqual to 7.55 Å. In the austenitic matrix not numerous M_6C carbides appear.

An extension of the aging time up to 50 hours entails an increase in the amount of carbides, these carbides occuring in the from of agglomerates /Fig12a,b and c/. As the aging time is prolonged up to 450 hours, the amout of the plate-like phase increases. As the electron diffraction analysis showed, a body centered cubic phase precipitates within the δ ferrite, its parameter being 7.10 Å. The plate-like precipitate within austenite had also a cubic lattice, but its parameter was markedly greater: a=8.65 Å /Fig.13a.b and c/. A thorough diffraction analysis, performed on thin foils, has shown that in ferrite a body-centered phase precipitates whereas in austenite the phase appears.

When the 0H17N14M2T steel was aged at 800°C, the occurrence of the σ phase in the high-temperature δ ferrite was ascertained already after 10 hours. After 50 hours very numerous precipitated particles of the $M_{23}C_6$ carbides appear /Fig.14a,b and c/. An extansion of the aging time up to 150 hours led to the precipitation of many phases with a lath-shaped and plate-like morphology /Fig.15a,b and c/. By applying diffraction analysis it was possible to show that the plate-like precipitate particles are surrounded by ferrite, their lattice being tetragonal. By measuring the lattice parameters the plate-like precipitate was identified to be the σ phase. The investigation made after 450 hours showed that the plates of the σ phase are no longer surrounded by ferrite but by austenite.



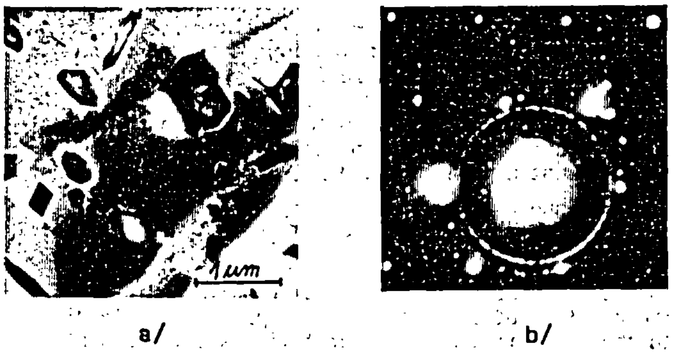
a/ b/



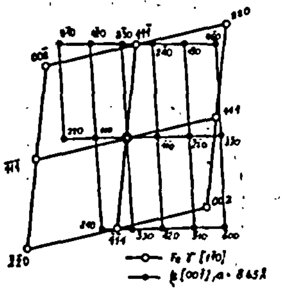
c/

Fig.12. 00H17N14M2 steel, quenched and aged: temperature 800°C, time 50 hours: a/ M_6C carbide precipitations isolated on an extraction replica; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.

When the 00H17N14M2 steel was aged at 900°C, isolated

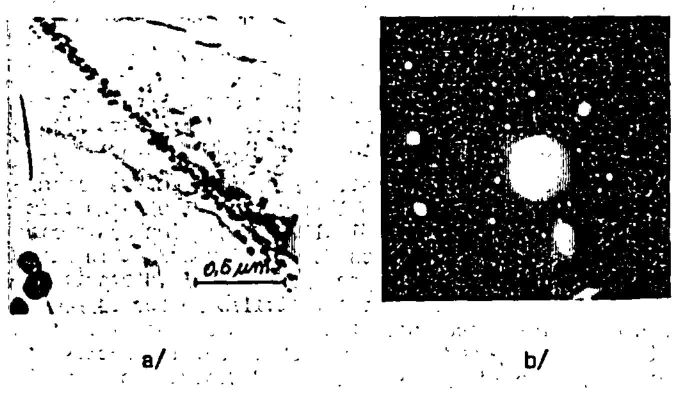


a/ b/

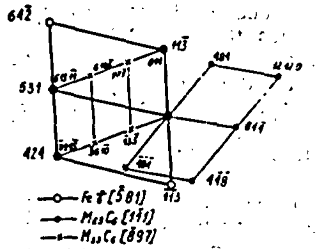


c/

Fig.13. 00H17N14M2 steel, quenched and aged: temperature 800°C, time 450 hours: a/ intermetallic χ phase procipitate; b/ electron diffraction pattern of χ and matrix; c/ solution of the electron diffraction pattern.



a/ b/



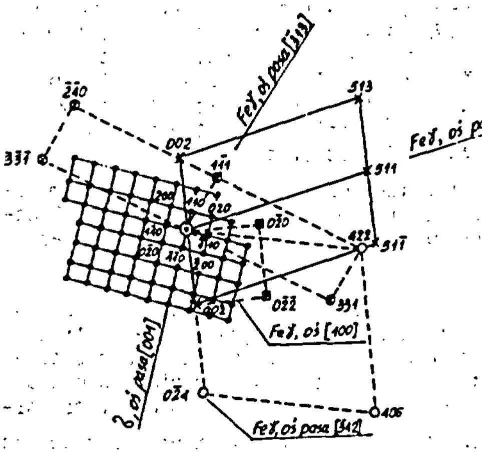
c/

Fig.14. 0H17N12M2T steel, quenched and aged: temperature 800°C, time 50 hours: a/ carbide precipitate at grain boundaries of austenite; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.

particles of a phase with the parameter a=7.54 Å as well as particles of the $M_{23}C_6$ carbides were found on extraction replicas already after 10 hours. If aging times were longer, the examination of this foils showed that in the neighbourhood of the δ ferrite a cubic body-centered phase with the parameter of about 8.0 Å occurs /Fig.16a,b and c/. Within austenite and at its boundaries the χ phase is present. The {411} planes of the χ phase are parallel to the {111} planes of austenite and have the same spacing.

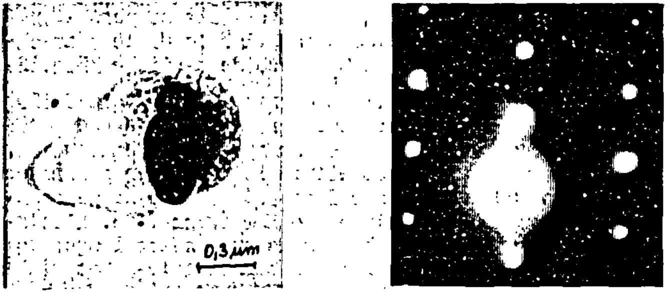
When the titanium-stabilized steel is being aged at 900°C, very numerous minute carbides appear alredy after 2 hours. After 450 hours, the carbide particles also occur at twin boundaries while within the austenite grains, apart from spheroidal carbides, numerous minute laths of the σ phase are seen.

a/ b/

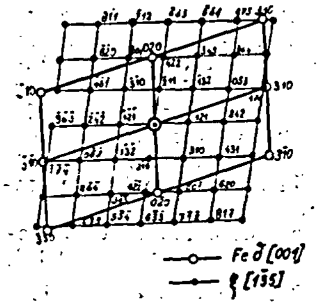


c/

Fig.15. 0H17N12M2T steel, quenched and aged: temperature 800°C, time 150 hours: a/ σ phase precipitate in α' ferrite; b/ electron diffraction pattern; c/ solution of the electron diffraction pattern.



a/ b/



c/

Fig.16. 00H17N14M2 steel, quenched and aged: temperature 900°C, time 450 hours: a/ intermetallic ξ phase precipitate in α' ferrite; b/ electron diffraction pattern of the precipitate and matrix; c/ solution of the electron diffraction pattern.

5. Discussion of the results

The investigation performed in the framework of the present work have proved that in austenitic chromium-nickel -molybdenum steels of the 17-14-2 type, in the course of aging, complex processes, connected with the precipitation of interstitial and intermetallic phases, take place. The evolution of this processes in titanium-stabilized steels differs from that in non-stabilized steels. Of essential importance for the development of the relevant processes in both steels is the presence of the high-temperature δ ferrite and - in the case of the 0H17N12M2T steels - also the occurrence of the MC carbides.

During the first two hours of the aging treatment at 600^{0}C, applied to the 00H17N14M2 steel, almost all excess carbon precipitates out of austenite in the form of interstitial phases. After this time, the concentration, in austenite, of metallic elements with great atomic diameters undergoes a change. These changes are not observable in the structure of the steel; they can, however, be ascertained by measuring the lattice parameter of austenite /Fig.7/. After longer aging times, intermetallic phases σ , χ and a new phase, designated by the authors as ξ , appear. The occurrence of the σ phase is strictly connected with the transformation of the high-temperature δ ferrite while the phases χ and ξ do not show any preferred precipitation sites.

At 700^{0}C, transformations of the high-temperature δ ferrite take place more rapidly; the phases χ and ξ occur and particles of the M_6C carbide begin to appear /Fig.3a and b/.

At 800^{0}C, the evolution of the precipitation processes turned out to be the fastest. Changes in the lattice parameter after two hours as well as after longer times are the greatest /Fig.7/. After aging during 10 hours the structure image is similar to that found after aging at 700^{0}C during 50 hours. Already after aging during two hours, apart from the $M_{23}C_6$ carbides, traces of the M_6C carbides were found. Essential changes take place in the process of transformation of the δ ferrite. Instead of the σ phase and the $M_{23}C_6$ carbides, plates of the cubic ξ phase, with the structure similar to that of the χ phase but having a smaller lattice parameter, occur. An extension of the aging time leads to a considerable coarsening of the M_6C carbide particle /Fig.12a, b and c/. The investigations carried out on thin foils have corroborated the presence of the χ phase and the body-centered cubic ξ phase. It has been found that the ξ phase precipitates within ferrite, its parameter varying in the broad range from 6.4 to 7.95 Å, depending on the aging temperature and the aging time. Electron diffraction analysis has shown that after longer aging times its lattice parameter approximates that of the χ phase and that this phase occurs surrounded by austenite /Fig,13a,b and c/. These circumstances indicate that the ξ phase is a transient phase. In the course of time the lattice parameter of this phase increases and the phase turns into the stable ξ phase. The ferrite, however, out of which the precipitation takes place, becoming more and more depleted in ferrite-forming elements, transforms into austenite. Independently of the observed decomposition of the δ ferrite, connected with the precipitation of the χ phase, the process of precipiattion of the χ phase within the austenite grains simultaneously take place.

Changes in the parameter of the austenite lattice, occurring in the 00H17N14M2 steel during aging at 900^{0}C, proceed more rapidly than at 700^{0}C but they are a little slower at 800^{0}C. It testifies to a slower evolution of the aging process. The degree of dispersion of the precipitated phases is, after aging at 900^{0}C, lower, i.e. the particles have a greater size /Fig.5a and b/. Electron diffraction analysis, performed on extracted particles, has shown that after 10 hours in the structure of the 00H17N14M2 steel the $M_{23}C_6$ carbides and the phase ξ are present /Fig.16a,b and c/. Electron diffraction studies, made on thin foils, showed that - after longer aging times - within austenite as well as at its boundaries plate-like particles of the χ phase occur while in the region of the δ ferrite particles of the ξ phase are found.

An analysis of the results of electron diffraction studies of thin foils, based on the stereographic projection, has shown that between the particles of the χ phase and austenite a repeated parallelis of crystallographic planes $(111)_{Fe_\gamma} \parallel (110)_\chi$ and directions $\langle 011 \rangle_{Fe_\gamma} \parallel \langle 110 \rangle_\chi$ exists.

On the basis of the experimental results a diagram of the kinetics of initiation of the processes - leading to the precipitation of interstitial and intermetallic phases - for the 00H17N14M2 steel has been elaborated /Fig.17/.

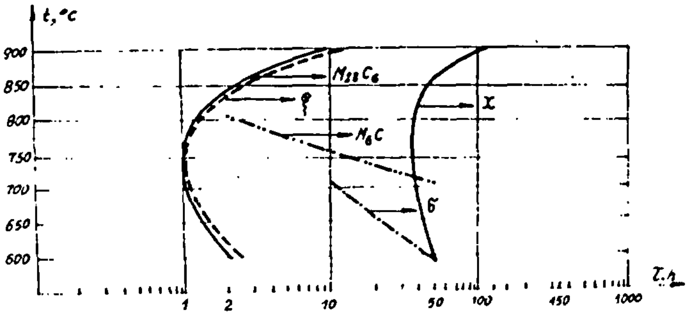


Fig.17. Effect of the aging temperature and aging time on the initiation of the precipitation of interstitial and intermetallic phases in the 00H17N14M2 steel.

The evolution of precipitation processes in the titanium-stabilized austenitic 0H17N12M2T steel differs from that presented for the 00H17N14M2 steel. The austenite lattice parameter decreases during 50 hours, this being the evidence of a slower precipitation of interstitial phases. Precipitation of intermetallic phases is accompanied by the appearance of the α' phase; precipitation of the phases χ and ξ was not observed. Indirectly, the view represented by L.K.Singhal and J.W.Martin as well as by other authors has been corroborated, according to which the σ phase nucleates within the metastable α' ferrite, formed around the earlier precipitated carbides of the MC type. The suggestion of the above authors as to the decomposition of the high-temperature δ ferrite has not been, however, corroborated; in every region of the σ phase which was examined, precipitated particles of the α' phase were found.

The investigations performed in the framework of the present work did not corroborate the results obtained by other authors, concerning the occurrence of the intermetallic Laves phase η . It may only be supposed that this phase is likely to appear in the steels in question after longer aging times than those applied in the present work.

6. Conclusions

On the basis of the results of investigations carried out on quenched and aged 00H17N14M2 and 0H17N12M2T steels the following conclusions can be formulated:

1/ In the 00H17N14M2 steel, at temperatures in the range of 600-900^{0}C, precipitation of the $M_{23}C_6$ carbides, of the body-centered cubic transient phase ξ and of the intermetallic χ phase takes place. In the temperature range of 600-700^{0}C the intermetallic σ phase appears within the high-temperature δ ferrite while at temperatures above 700^{0}C the transient ξ phase precipitates. Independently of the phases precipitated in the temperature range of 700-800^{0}C, M_6C carbides also precipitate.

At first, precipitation of the $M_{23}C_6$ carbides and of the transient phase ξ takes place; after longer aging times, depending on the temperature, the intermetallic σ phase, the M_6C carbides and the χ phase precipitate.

2/ In the OH17N12M2T steel, carbides of the MC and $M_{23}C_6$ type as well as the intermetallic σ phase precipitate during aging. Precipitation of the σ phase takes place at first within the high-temperature δ ferrite while after longer aging times it precipitates within austenite in the form of laths and - probably via the metastable α' ferrite - in the form of plates.

The authors of the present work express their thanks to Mr Sławomir Pilarczyk, Eng., to Mrs Maria Bukowska, Dr Eng., and to Mr Jerzy Jakubowski, M.Eng. for their help in carrying out the electron and X-ray diffraction analyses.

CONCRETE DEGRADATION MONITORING & EVALUATION

Narendra Prasad and Richard Orr

Summary

Concrete structures of nuclear power plants are potentially subject to age-related deterioration mechanisms. These plants are generally licensed for a term of 40 years. In order to maximize the return from the existing plants, feasibility studies for continued operation of these plants beyond the original licensed life span are required.

This paper describes the degradation mechanisms which may reduce the useful life of nuclear power plant structures and discusses the type of surveillance required for each plant. Recommendations are made for research and code committee action to develop industry position to address the key life extension issues.

Introduction

Concrete structures are extensively used in nuclear power plants due to inherent strength and durability. These structures are expected to crack in the regions where tensile stresses are developed. Crack widths are controlled by design to preclude severe degradation. The structures are subject to potential deterioration due to several environmental conditions, including weather exposure, ground water exposure, and high temperature and radiation levels. The nuclear power plants are generally licensed for a term of 40 years. In order to maximize the return from the existing plants, feasibility studies are in progress for continued operation of many of these plants beyond the original license. A study performed to evaluate degradation and to define appropriate condition monitoring and maintenance procedures for the nuclear power plant concrete structures is described in the following paragraphs. The study included concrete structures such as the containment buildings, interior structures, basemats, intake structures and cooling towers. Age-related deterioration at several operating power plants was surveyed and the potential degradation mechanisms have been identified. The elimination or control of one or more of these mechanisms can significantly inhibit the degradation process. This paper outlines a generic program to monitor the onset of the degradation processes, and describes various available repair techniques and their application. For plant-specific conditions, comprehensive condition monitoring and maintenance programs are required.

Critical Components and Degradation Manifestation

The first item in the scope of the study was the identification of the vulnerable components of the nuclear power plants. The following Category I structures are some of the critical concrete structures in nuclear power plants. It is necessary to demonstrate continued function for these structures under severe environmental and accident conditions.

- Containment Shell/Shield Building
- Containment Basemat
- Containment Interior Structures
- Auxiliary and Control Buildings
- Water Intake and Discharge Structures
- Cooling Towers

This study on the degradation of concrete structures included site inspections and a comprehensive review of published information on deterioration experiences in commercial nuclear power plants.

A visit to Shippingport, PA, where the U.S. Department of Energy is performing a nuclear reactor decommissioning operation, was included as part of this study. The prime objective was to gather data on the condition of the structures subjected to prolonged nuclear environment. The concrete structures are primarily below grade and were all in excellent condition. There were a few hairline shrinkage cracks on the interior faces. There were fine cracks on the top slab which was exposed to the outside atmosphere. The cracking appeared generally to follow the construction joints between pours. Shear keys and water stops had been used at these joints. The degree of cracking would not have prevented continued function of the concrete slab.

Cases of concrete degradation in commercial nuclear power plants in the United States are reported to the regulatory bodies and are published in "Nuclear Power Experiences".[1] These records of the several years were reviewed, together with records for other massive concrete structures similar to those in nuclear power plants, and generic issues were identified as follows:

Construction Deficiencies
- Voids in Concrete Due to Congestion of Rebars and/or Embedment
- Building Settlements

Prestressing System Degradation
- Tendon Grease Assembly Leakage
- Broken Tendons
- Cracked Anchors
- Anchor Head Failure

Concrete Surface Degradation
- Cracks
- Rust Stains
- Spalling
- Chemical Erosion

Foundation Degradation
- Basemat Cracking
- Turbine Generator Foundation Cracking

Degradation Mechanisms and Consequences

Age-related degradation mechanisms may produce accelerated aging if allowed to continue uninhibited for an extended period. However, these processes can be detected before any significant adverse effect has occurred. Potentially detrimental mechanisms are:

- o Alkali-Aggregate Reactions
- o Water Chemical Reactions
- o Environmental Conditions
- o Thermal Fatigue
- o Radioactivity

Nuclear power plant concrete structures are designed to be durable and are constructed under stringent quality control requirements to withstand extreme load environments which far exceed the normal operating conditions. Therefore, the above mentioned mechanisms would not be expected in nuclear power plant concrete structures. However, these mechanisms are to be considered in evaluating the degradation potential of a concrete structure.

Concrete is a highly complex material composed of a binding medium within which are embedded fine and coarse aggregate particles. Aggregates generally occupy 60% to 80% of the volume of concrete. The most commonly used aggregates are sand, gravel, and crushed stones. Expansive reactions between the active silica in the aggregates and the alkalis derived from cement hydration can cause concrete cracking.

For many concrete structures exposed to a moist environment, such as the groundwater or a flowing river or stream, the dissolved chemicals in the water present a potential degradation source. The undesirable chemicals are acids, sulfates and chlorides. The sulfates of sodium and potassium react with the hydrated lime and hydrated calcium aluminate in the cement paste. The calcium sulfate and calcium sulfoaluminate formation is associated with considerable expansion which disrupts the concrete.

Concrete structures suffer widespread deterioration due to aggressive environments such as acid rains, leaching and rebar corrosion. Properly designed, poured and cured concrete is generally impervious to water. However, certain environment, such as acid rains, may adversely affect even good quality concrete. In order for the chemicals to significantly attack concrete, they, in general, must be in a solution form and above a minimum concentration. The harmful chemicals are sulfates of sodium, potassium or magnesium. The products of combustion of fuels, which produce sulfurous gases, combine with moisture to form acids which attack concrete. Furthermore, water in contact with concrete, with cracks or improperly treated construction joints, may wash out calcium hydroxide and other solids, leading to disintegration of the concrete. The 'leaching' results from successive wetting and drying of the concrete. As the concrete dries, salts are deposited through evaporation of the water or interaction with carbon dioxide in the atmosphere. Extensive leaching causes an increase in concrete porosity, which leads to reduced strength and increased vulnerability to hostile environments. Evidence of leaching is in the form of disfiguring of concrete caused by deposits due to the solidification of the lime deposits on concrete surfaces. Concrete rebar corrosion occurs where concrete pH is reduced to less than 11. The buildup of corrosion products around steel rebars and other embedments tends to act as a wedge to split and spall the adjacent concrete. The corrosion process is aggravated by the presence of chlorides in seeping water. The degradation process can be detected by monitoring of cracks and can be controlled by sealing of the cracks. Corrosion can also be retarded by prevention of chlorides, water, oxygen or carbon dioxide from access to the rebars.

Concrete structures are also subject to freezing of the concrete while wet, followed by subsequent thawing. The internal hydraulic pressure created by the expanding ice-water system during freezing can result in concrete spalling. Repeated freeze/thaw cycling can lead to severe degradation.

The degradation of concrete due to a sustained radioactive environment has been studied. Petrographic examination found neither any recognizable visual degradation to the concrete core, nor any cracking in the matrix or of aggregate particles. The results indicated no degradation for concrete subjected to radiation levels similar to those experienced in the most severe locations in a nuclear plant.[2]

Condition Monitoring Techniques

The available techniques for monitoring the condition of concrete structures can be broadly classified in the following categories:

- o Visual Examination
- o Ultrasonic Tests
- o Other Non-destructive Examinations
- o Concrete Core Drilling
- o Groundwater Tests
- o Prestressing System Inspection

Visual examination is a frequently used procedure to identify areas of distress and may include a cracking survey and subsequent surface mapping. Photographs, including a scale to indicate linear dimensions, are of great value in condition monitoring. A cracking survey is performed to locate the cracks and to determine the size, type, and the relationship of the cracks with any other deterioration. A crack comparator can be used to determine the widths of cracks with an accuracy of 0.025 mm. The visual examination records evidence of scaling (area, depth, type), spalls and popouts (number, size, depth, type), rust stains, and exposed steel. Surface mapping includes preparation of permanent drawings from hand mapping, photographs, video films etc. These drawings show areas of cracking, spalling, scaling, honeycombing, popouts, exudation, erosion, unusual discoloration, seepage, conditions of joints and joint materials, rebar corrosion and soundness of surface concrete.

The various procedures for the ultrasonic tests, based on wave propogation principles, can be classified under the following categories:

Pulse-Velocity techniques
Pulse-Echo techniques
Impact-Echo techniques
Impact-Radar techniques

Pulse-Velocity testing is probably the most extensively used ultrasonic examination. The procedure involves the transmission of a longitudinal wave, by electro-mechanical means, in the concrete structure and measuring the time it takes to propogate through the thickness. The time between the transmitted and received pulses is measured to calculate the pulse velocity. The pulse velocity depends primarily on the elasticity and the density of the material. Changes in pulse velocity may indicate invisible cracks and voids.

In the Pulse-Echo technique, acoustic waves transmitted through concrete change directions when they encounter voids, cracks, rebars etc., and echo back to the transmitter. The reflected waves are converted to electrical pulses and displayed on oscilloscopes to show the intensity and transit time.

The Impact-Echo method employs spherical stress waves using a mechanical impact source. The pulse propogates through the object and is reflected by material flaws. The reflected signals are analyzed on an oscilloscope. The technique has been limited to laboratory studies and has had very little field application.

In Impact-Radar approach, low-power electromagnetic pulses radiated into concrete are reflected back to the transmitting/receiving antenna and are recorded. The transmission time and the amplitudes are measured to detect the voids, rebars, cracks, and other foreign material in concrete.

Other non-destructive examination techniques used for concrete include the Windsor Probe, Rebound Hammer, and Cast-in Place Pullouts. These techniques are extensively used for testing aging concrete structures. Standard ASTM test specifications are available for these test methods.

For certain applications such as massive concrete foundations, the techniques discussed above may not provide conclusive information. Core drilling is an accepted procedure for areas which otherwise are inaccessible for examination. Detailed procedure and guidelines are provided in ACI Standard ACI 207.3R, and ASTM specification C-823. ASTM procedures are also available for determining the physical properties of drilled concrete cores.

Groundwater testing provides information about the potential harmful effects of the dissolved chemicals in the groundwater which may be in contact with concrete. An early awareness allows timely action to preclude the possibility of ongoing deterioration. Groundwater samples are tested to monitor the pH value, chloride and sulfate contents etc. The water chemistry should be documented twice a year to account for the seasonal changes.

Potential deterioration of prestressing systems includes loss of prestress in tendons, and cracks in wires and in anchor blocks. Stringent in-service monitoring procedures are essential. For cost-effectiveness, it is imperative to control the environment of the tendon anchor hardware to eliminate hydrogen stress cracking. Anchor heads are subjected to visual examination to reveal the presence of water in the grease caps. In some cases magnetic particle testing is required to detect the cracks. If there is any evidence of cracks, detailed metallurgical evaluation is needed.

Acceptance Criteria

To evaluate the test results, there is a need to define appropriate acceptance standards. For example, should we look for and map hair-line cracks and take preventive measure at first sight of these cracks, or should we wait until there is evidence of leaching and rust. General guidelines for tolerable crack widths for typical environments for nuclear power plant structures have to be established. The following table, based on reference 3, provides guidelines.

Environmental Condition	Tolerable Crack Width (mm)
Water retaining structures	0.10
Seawater exposure, wetting and drying	0.15
De-icing chemicals	0.18
Humidity, moist air, soil	0.30
Dry air or protective membrane	0.41

These guidelines have to be used in conjunction with additional structural data, such as the type (prestressed or reinforced) and size of the structure, concrete cover over the rebars, etc. In some cases wider cracks may be acceptable.

The condition and the functional capability of the components of prestressing systems are evaluated in accordance with the provisions in Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures." The inspection program, based on this regulatory guide, provides reasonable assurance that the structural integrity of the component is maintained.

Limits are also established for other forms of degradation, such as the erosion, corrosion, spalling, thermal fatigue, groundwater chemistry, etc. As a guideline in evaluation of groundwater chemistry, a pH value of 5.5, or chloride and sulfate contents greater than 500 and 1500 ppm, respectively, warrant further evaluation to minimize progressive deterioration. The acceptable limits for other forms of degradation, such as rust staining due to rebar corrosion, concrete spalling, concrete distress, etc., are set for various structures on a case by case basis taking into consideration the site-specific conditions.

Degradation Evaluation Program

The following steps are recommended for development of a plant-specific degradation evaluation program. These plans should include detailed inspection/monitoring procedures, evaluation criteria and inspection frequencies.

- Review of Plant Records
  - o Structural Drawings
  - o Construction Materials Used (Type of cement, aggregate, admixtures, rebars. etc.)
  - o Results of Structural Acceptance Test and Integrated Leak Tests
  - o Groundwater Records (water chemistry, seasonal level variation, sump pump records, etc.)
  - o Atmospheric Conditions (freeze/thaw cycles, acid rains, etc.)
- Identification of Structures with Degradation Potential
- Identification of Possible Degradation Mechanisms
- Development of Monitoring Program for Identified Structures
  - o Appropriate Tests Identification (surface mapping, ultrasonic tests, groundwater tests, etc.)

- o Definition of Indications to be recorded and Acceptable Limits of Deterioration (crack size, erosion, chemical contents in water, etc.)
- o Computerized Documentation of Test Results
- o Testing Frequency Specification

Repair and Restoration

The objective of the restoration process is to arrest deterioration, restore deficient concrete, prevent leakage and assure structural integrity. A variety of techniques are available for repair of massive concrete structures, such as bridges, dams and embankments. While most of these are common repair practices, some patented commercial methods are used for specialized jobs. These procedures can be conveniently adapted for nuclear power plant concrete structures. For specific application, a selected procedure should take into consideration the cause of distress, environmental conditions, cost/benefit ratio, and should fulfill the purpose of repair, such as to provide water tightness, prevent access of corrosive materials to rebars, or to enhance functional performance.

Repair materials used are conventional cement mortars, mortars utilizing latex, epoxy resins, or methyl methacrylate, and, in some cases, certain chemical monomers which polymerize to provide the necessary filler material and bond. Methods commonly used for crack repairs are epoxy injection, routing and sealing, grouting and polymer impregnation. Epoxy injection is preferable only when the crack surface is dry. Cracks as narrow as 0.05 mm can be bonded by the injection of epoxy. Routing and sealing consists of enlarging the cracks and filling them with an appropriate sealant. Chemical grouting is effective when water begins to flow through a crack or joint. It requires injection of a medium into the channel through which water is flowing so that the secondary substance permanently fills the channel. Surface impregnation and polymerization involves flooding the dry surface with a monomer, and subsequent polymerization in place. The cracks are mended when the monomer is polymerized. The repair of honeycombs, voids, and holes around congested reinforcement is performed by an application of either a fine grain cement concrete (cement mortar) or an epoxy fine grain concrete. Gunite fine grain concrete is considered the most reliable method for repairing voids. Scaling is generally caused by generation of internal pressure during freezing of the solution contained in the voids. Surface coatings (less than 6 mm thick) are used when the scaling is in its early stages. For greater thicknesses, various kinds of overlays, such as fiber reinforced concrete, latex modified concrete, or polymer impregnated concrete systems, are recommended.

Codes and Standards

Currently, there are no mandatory requirements for periodic inspection of nuclear power plant concrete structures. The only inservice requirement is for periodic leakage rate tests for the containment. The inspection of other safety related concrete structures, which may be critical to nuclear plant life extension, is not covered by any nuclear code or standard. There is a need to develop industry codes to accommodate plant life extension issues either by modifying existing codes or by issuing new codes. Items to be considered for concrete structures would include:

- o Scope and Frequency of Condition Survey
- o Non-Destructive Examinations
- o Acceptance Standards
- o Maintenance and Restoration Procedures

The major initiative on codes and standards related to nuclear power plant life extension is being sponsored by the Nuclear Plant Life Extension Codes and Standards Subcommittee. This committee, known as NUPLEX, was developed for the purpose of organizing input from ACI, NRC, ASME, etc. The principal objective is to ensure the development and/or revision of codes and standards to support and facilitate the nuclear plant life extension. The American Concrete Institute (ACI), which is responsible for the ACI Standard 349, "Code Requirements for Nuclear Safety Related Concrete Structures," is represented on this committee.

Research and Development

Additional research is required to support the extension of nuclear power plant concrete structures. Industry development of code requirements was discussed in the previous section. Such code requirements must be based on a full understanding of the potential deterioration mechanisms and their rate of growth. As an example, concrete is designed to crack; if the cracks are excessive, deterioration will accelerate due to ambient environmental conditions. Research studies are required to determine the acceptance criteria for observed cracking and the stage at which repairs should be intiated.

The frequency of containment testing should be evaluated to determine if this testing is causing unnecessary deterioration. A reinforced concrete containment is subjected to internal pressurization for leak rate tests. During the Structural Acceptance Test (SAT), the containment is required to withstand an internal pressure 15% higher than the design pressure. The Integrated Leak Rate Tests (Type A) are conducted at a pressure which subjects the containment to stresses smaller than those of the SAT. Three such tests are to be performed at approximately equal intervals during each 10 year service period. The ILRT re-opens existing cracks in the concrete, and the repeated opening may create a permanent seepage path for water to attack rebars. A decrease in the frequency of Type A test would enhance the continued operation prospects of the containment without diminishing the value provided by the tests. Significant data on past ILRTs is available, and there is a potential to relax the requirements. A program should be developed to demonstrate that the ILRT could be performed less frequently than the current requirement.

Concrete in nuclear power plants is exposed to long term thermal and radiation environments. The long-term deformation of the concrete due to shrinkage and high temperature may initiate cracks and cause embrittlement. Although some test data is available, additional investigation may be useful to provide conclusive evidence about ineffectivity of these mechanisms. Rebound hammer test, ultrasonic non-destructive examination and, if possible, concrete core drilling and testing can provide substantive information. The nuclear industry should implement a comprehensive research program to resolve these issues.

Conclusion

The vulnerability of nuclear power plant concrete structures to potential degradation mechanisms has been evaluated. The deterioration processes are detectable and site-specific monitoring programs should be considered for application. Remedial measures, if required, are available to protect the structures from these degradation mechanisms. There is a need to incorporate changes in ACI Standard 349 to address these issues related to the nuclear plant concrete structures.

References

1. "Nuclear Power Experiences," Vol. PWR-2 published by Nuclear Power Experiences, Inc.

2. Buck, A. D., "Characterization of Radioactive Concrete by Petrographic and Physical Methods," ACI Materials Journal, January-February 1988.

3. ACI Committee 224, "Control of Cracking in Concrete Structures," (ACI 224R-80), American Concrete Institute, Detroit, 1980.

4. Naus, D. J., "Regulatory Analysis of Regulatory Guide 1.35 - In-service Inspection of Ungrouted Tendons in Prestressed Concrete Containments," NUREG/CR-4712, February 1987.

5. ACI Committee 224, "Controlling of Cracking in Concrete Structures," (ACI 224R-80), American Concrete Institute, Detroit, 1980.

6. ACI SP-85, "Rehabilitation, Renovation and Preservation of Concrete and Masonry Structures," American Concrete Institute, Detroit, 1985.

7. ACI SP-82, "In Situ/Nondestructive Testing of Concrete," American Concrete Institute, Detroit, 1984.

THERMAL AGING BEHAVIOR OF MARTENSITIC STAINLESS STEEL

M.TSUBOTA, K.TAJIMA, K.HATTORI AND H.KASHIWAYA

TOSHIBA Corporation

ABSTRACT

13Cr-4Ni-Mo cast martensitic stainless steel, which is used as a material for pumps in LWR (Light Water Reactor) plants, was aged at 350°C and 400°C for up to 10,000 hours and the effects of heat treatment and alloy composition on the toughness and SCC properties of the alloy were examined.

Toughness of the alloy evaluated by Charpy impact test at 0°C decreased with aging time, and it was clarified that molybdenum addition suppressed the degradation of the alloy during the aging. From the Arhennius relations in data of the impact test, the activation energy for the toughness change was estimated to be around 40 Kcal/mol, and it was predicted that the alloy with more than 0.3% Mo could be used for 40 years in BWR environment.

The interaction between molybdenum and phosphorus acts an important role in the thermal degradation of 13Cr-4Ni-Mo martensitic stainless steel. From the AES and TEM analysis, it was speculated that two different mechanisms for toughness loss acted simultaneously during aging. One was precipitation of fine particles, which hardened the matrix of the alloy and was accelerated by molybdenum addition. The other was segregation of phosphorus at the grain boundary, which weakened the grain boundary toughness. Although the latter phenomenon was dominant in the alloy investigated, the effect of grain-interior embrittlement by precipitation hardening should be considered and the level of molybdenum content should be limited.

SCC susceptibility was evaluated by the creviced bent beam test in simulated BWR water. SCC occurred intergranularly and slightly enhanced by the aging. But, compared to the other unaged structural materials used in LWR such as sensitized 304 stainless steel and alloy X-750, SCC susceptibility of the alloy was considerably small and thought to be negligible, even after a long time aging.

Heat treatment prior to aging also affected the property of the alloy and it was proved that, from the view point of thermal degradation, the heat treatment with lower temperature normalizing followed by double-tempering may be the appropriate heat treatment for the alloy.

INTRODUCTION

From the view point of plant life extention (PLEX), thermal degradation of the alloys used in the light water reactor is one of the crucial factors which decide the plant life, and understanding the effects of thermal aging on the material characteristics is an important matter for performing the PLEX.

Since 1982, in order to investigate the applicability of 13Cr-4Ni martensitic stainless steel as a structural material for BWR, Toshiba has been carrying on the study concerning the thermal aging of the alloy (1).

13Cr-4Ni martensitic stainless steel has been developed as a material for the impeller of water turbines or pumps. This material is also used in some light water reactors (LWRs) as the blades of pumps because of its high strength and good resistance to corrosion.

There were many studies on the degradation of ferritic or martensitic alloy steels in high temperature service (2,3,4), and it has been shown that many alloys had possibility of degradation after the long time operation at the elevated temperature. But, inspite of the importance of knowing the mechanical and corrosion properties of the alloys used in LWR environment, there were few published work concerned with the property change after the long time use under LWR environment.

Therefore, in this study, the effect of molybdenum content, which is usually added to the alloy for improving the temper-embrittlement and toughness at the low temperature (10,11), and heat treatment condition on the degradation of the alloy were investigated.

EXPERIMENTAL PROCEDURE

Material

Four cast 13Cr-4Ni alloys have been prepared for the examination. Three alloys with various molybdenum content (M0, M3 and M6) were melted by VIF (Vacuum Induction Furnace) and the other one (HP) was melted by ESR (Electro Slag Remelting). Chemical compositions of the alloys are described in Table 1.

They were cast into the mold and processed the following heat treatments.

HT I : 1050°C/5h + 1020°C/1h + 640°C/1h + 620°C/1h
HT II: 990°C/4h + 600°C/4h + 580°C/4h

Table 1 Chemical composition of the alloys (wt%).

Alloy	C	Si	Mn	P	S	Cr	Ni	Mo
M0	0.043	0.36	0.59	0.023	0.002	11.90	3.90	0.01
M3	0.050	0.43	0.63	0.019	0.004	12.30	4.16	0.29
M6	0.053	0.41	0.61	0.022	0.003	11.95	4.17	0.57
HP	0.035	0.48	0.71	0.034	0.015	12.35	4.10	0.41

Both of them are the heat treatments specified in the company specification for 13Cr-4Ni alloy.

Alloy M0, M3 and M6 were heat treated with HT I condition for examining the effect of molybdenum content, and alloy HP was heat treated with both HT I and HT II conditions for examining the effect of heat treatment.

Cast blocks were cut into several small pieces and aged at 400°C and 350°C for up to 10,000 hours. Test specimens were machined from the small blocks taken from the furnace after the various periodic time of aging.

Charpy Impact Test

Toughness change after long time aging were checked by Charpy impact test at 0°C. The configuration of the Charpy impact specimens was 10mm x 10mm x 55mm with 2mm depth V-shape notch.

Hardness

Some chromium stainless steels, which are similar to the alloy investigated in this study, have a possibility of age-hardening by the precipitates produced during the aging (5,12). Hardness test is a convenient method for detecting the microstructural change in grain interior. Therefore, hardness was measured by a micro-Vickers hardness tester(1Kg).

SCC Test

The CBB (Creviced Bent Beam) technique (13) was employed for the SCC test. As shown in Fig. 1, specimen (20mm(w) x 50mm(l) x 2mm(t)) and graphite wool (SIGRI KFB-2), which is crevice former, were sandwiched by the devices and preserved into the autoclaves filled with circulated high temperature pure water which contained 20ppm O_2.

The tests were carried out at 288°C for 500 hours, and the SCC susceptibility was evaluated by crack depth measurement on a longitudinal section of the specimen.

Auger Electron Spectroscopy

Since it was thought, from the many experiences in turbine materials, that the loss of toughness by a long term aging was mainly caused by phosphorus segregation, Auger Electron Spectroscopy (AES) analysis has been performed to check the segregation of impurities and other elements at the grain boundary of aged and unaged specimens. Specimens for AES analysis were chilled by liquid nitorogen and fractured in the vacuum chamber to prevent the contamination. AES observation was operated by ULVAC-PHI 600 under 10^{-10} torr.

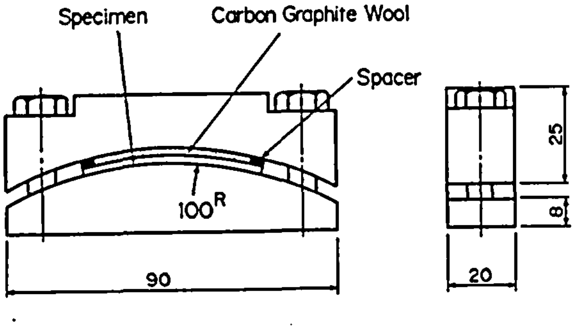


Figure 1. Schematic view of the CBB test device.

The microstructural change and fracture surfaces of the specimens were also observed by STEM/EDS and SEM.

RESULTS

Charpy impact test

Figures 2(a) and (b) show the change of the Charpy impact energy of the alloys M0, M3 and M6 isothermally aged at 400 and 350°C, respectively. Although the toughness of each specimen decreased with aging time, higher molybdenum content delayed the impact energy decreasing.

From the Arhennius relations in the data of Charpy impact energy for alloy M0, activation energy for the toughness change has been estimated to be 37-39 Kcal/mol, which is very similar to the value for the phosphorus segregation in iron described in other investigations (14).

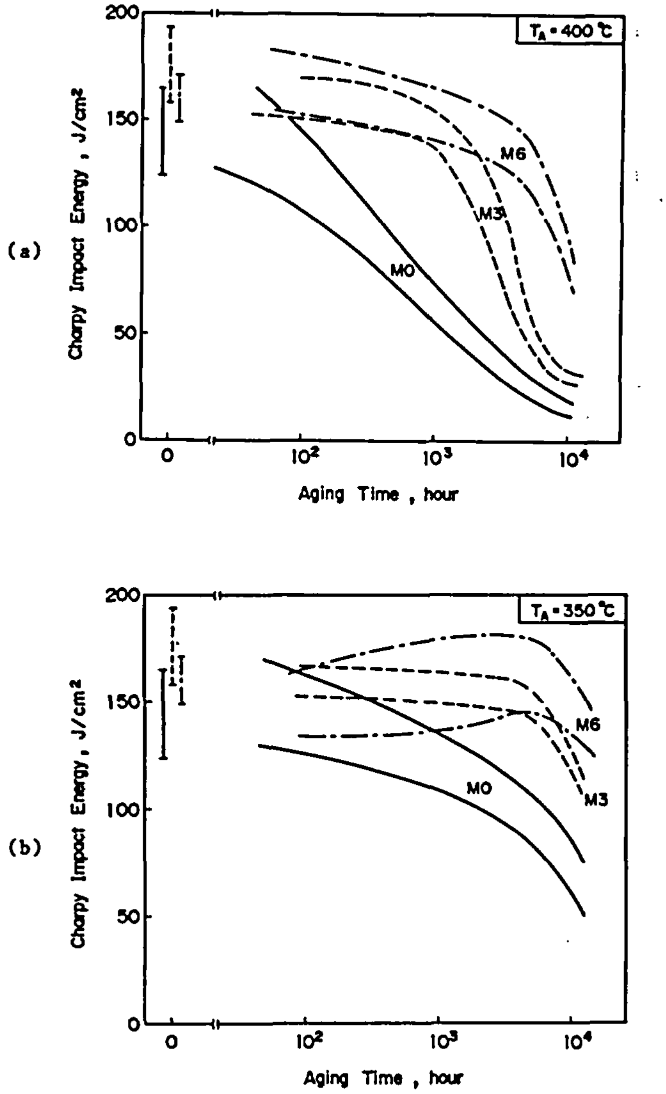


Figure 2. Variation in Charpy impact energy for the specimens M0, M3 and M6 aged at (a) 400°C and (b) 350°C (1).

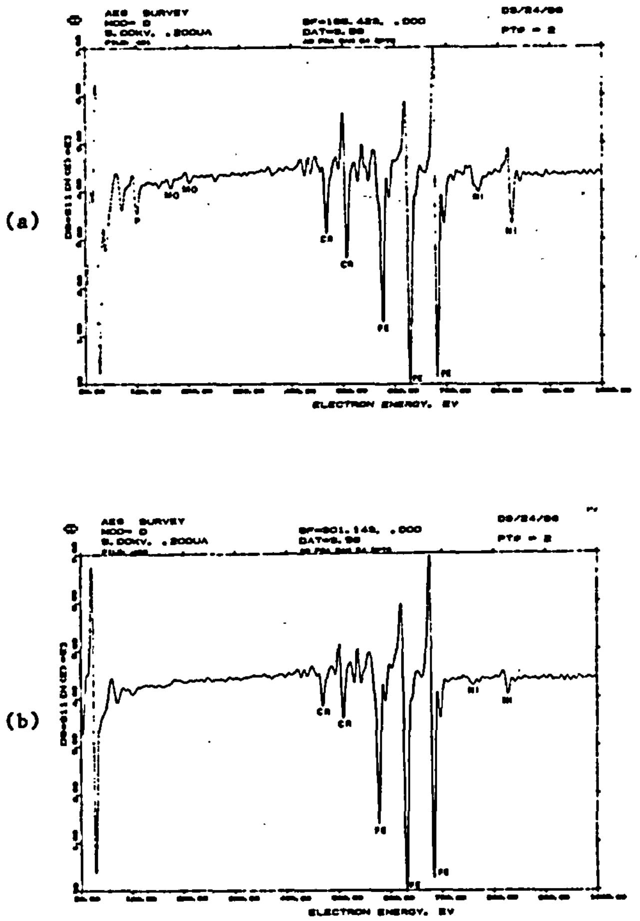


Figure 3. Examples of AES energy charts: (a) grain boundary and (b) grain interior of alloy M6 aged at 400°C for 10,000 hr (1). Distinctive peaks for phosphorus and molybdenum appeared in chart (a).

By AES analysis, it was proved that the intergranular fracture surfaces were covered with phosphorus, as shown in Fig. 3. The quantitative calculation based on the equation described in (15) showed the tendency of increase in phosphorus concentration at the grain boundary by aging (Fig. 4), and also showed that higher molybdenum content made less concentration in phosphorus at the grain boundary after aging (Fig. 5).

The tendency in the concentration of phosphorus at grain boundaries was in good agreement with the change in toughness shown in Fig. 2. Consequently, it has been concluded in (1) that the phosphorus segregation at grain boundaries might be a main reason for the loss of toughness in 13Cr-4Ni-Mo alloy and addition of molybdenum is necessary to delay the degradation in toughness.

Using the activation energy obtained and aging-parameter (P) (12), the prediction of the impact energy of the alloy after long time service under the BWR temperature 288°C was made (Fig. 6)(1). After 40 years usage at 288°C, the impact energy of alloys with 0.3 and 0.6% molybdenum do not change so much, hence, the impact energy of the alloy without molybdenum decreases to a half value.

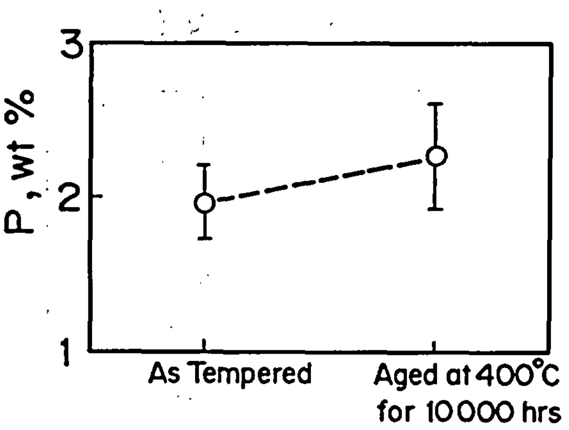


Figure 4. Phosphorus concentration at the grain boundary of alloy MO in as-tempered and as-aged conditions (1).

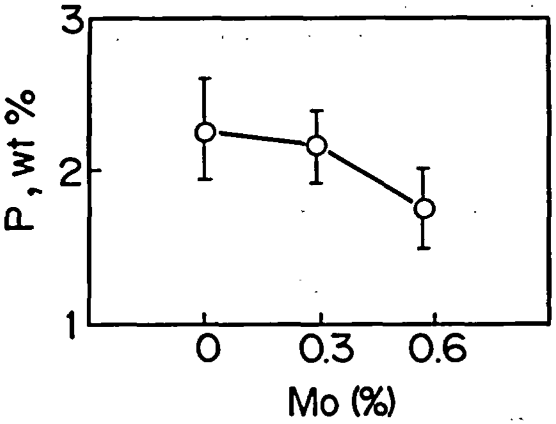


Figure 5. Phosphorus concentration at the grain boundary of alloy MO, M3 and M6 after aging at 400°C for 10,000hr (1).

On the basis of activation energy of around 40 kcal/mol, the results above suggest that addition of molybdenum, which must be more than 0.3% in amount, can make 13Cr-4Ni-Mo steel to prevent the degradation in toughness during service life of BWR.

Figure 7 is the result of toughness change in alloy HP. From this result, it can be easily concluded that the heat treatment prior to aging is another factor which affects the degradation characteristics, and HT II heat treatment is thought to be more appropriate for the alloy to delay a degradation in toughness than HT I.

Hardness

As shown in Fig. 8 and 9, hardness in every specimen increased with aging time, and it was conjectured that some isothermal microstructural change occurred in the matrix of the alloy.

Figure 10 is a TEM photograph of the 400°C/10,000 h aged alloy M6. There can be seen small particles in the matrix. EDS and electron beam diffraction pattern analyses were tried, but it has not been clarified what they were.

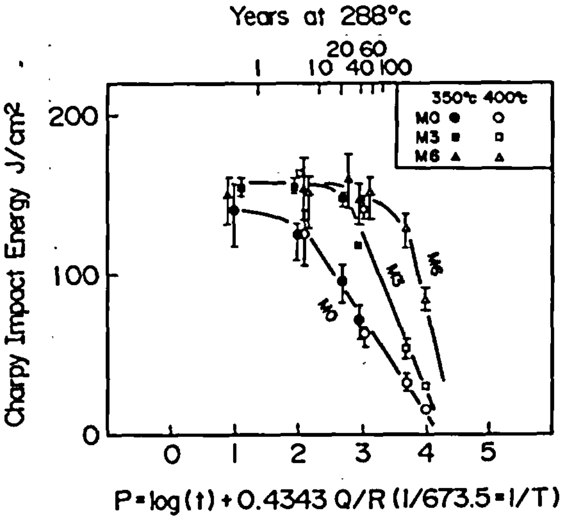


Figure 6. Prediction of Charpy impact energy of the alloys M0, M3 and M6 aged at BWR temperature 288°C. Activation energy Q for the calculation is 40 kcal/mole estimated from the toughness change in alloy M0 (1).

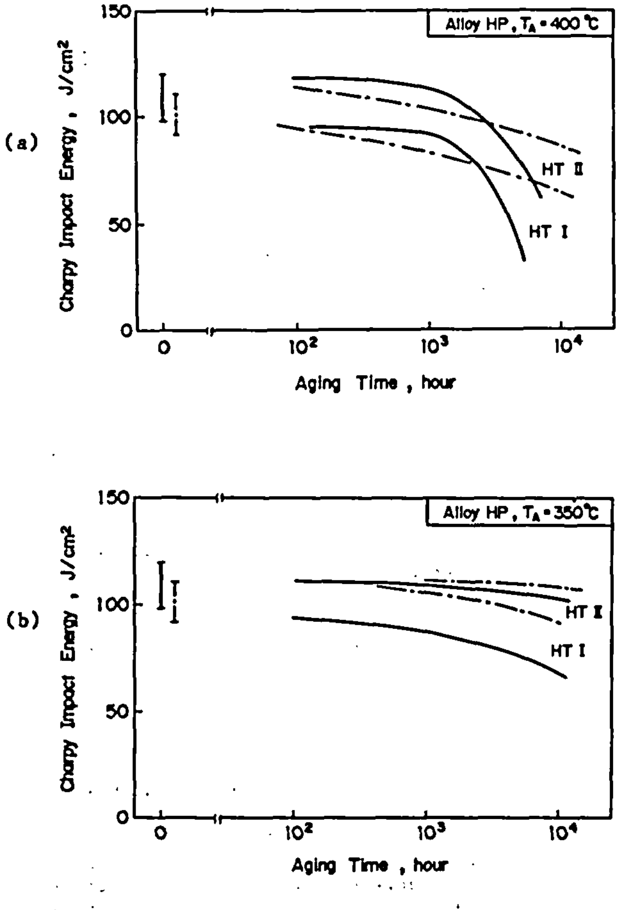


Figure 7. Variation in Charpy impact energy for the specimen HP aged at (a) 400°C and (b) 350°C.

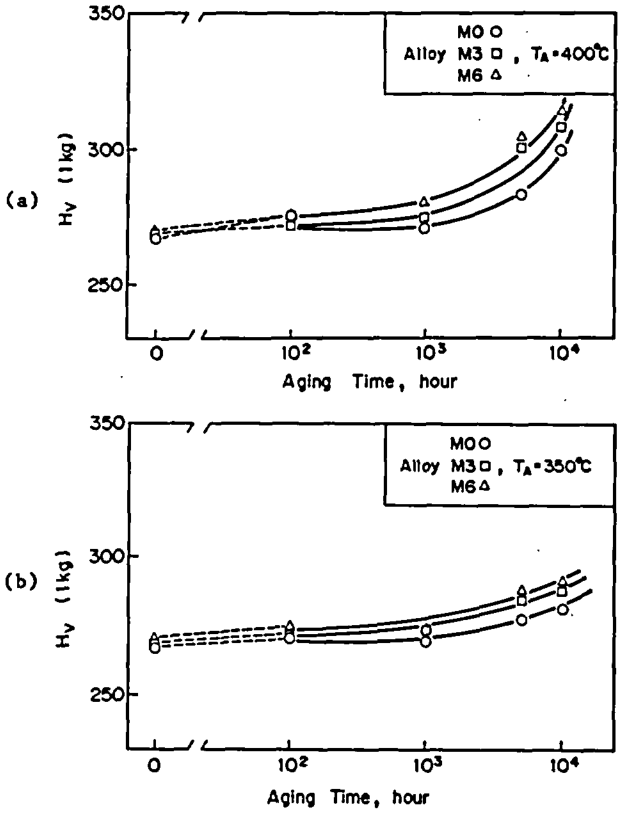


Figure 8. Hardness for the alloys M0, M3 and M6 after aging at 400°C and 350°C (1).

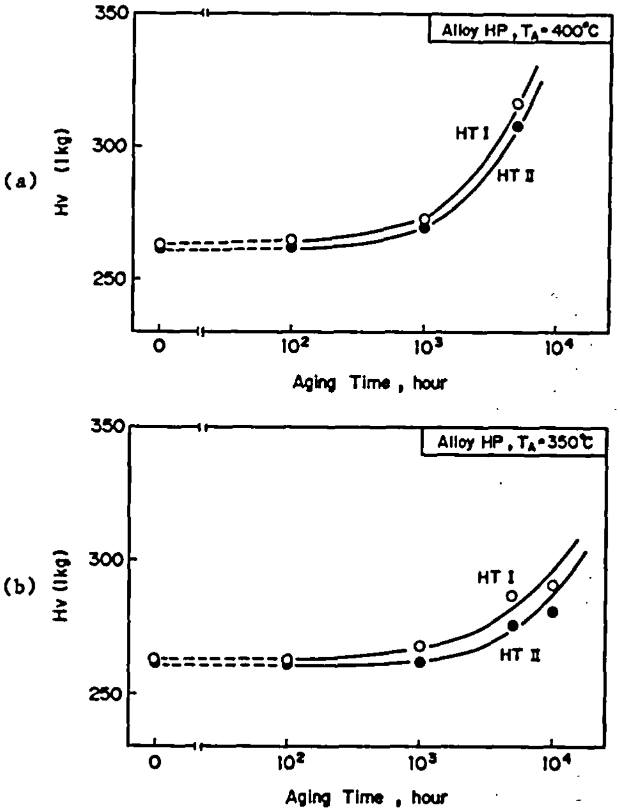


Figure 9. Hardness for the alloys HP after aging at 400°C and 350°C.

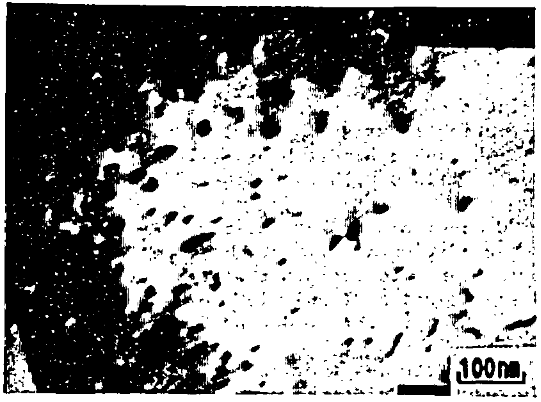


Figure 10. TEM photogragh for precipitates observed in 450°C/10,000hr aged M6 (1).

From the results of hardness tests and TEM observation in alloy M0, M3 and M6, it has been supposed that these precipitates, which might have hardened the alloy, formed independent of the existence of molybdenum, but its precipitation was accelerated by the existence of molybdenum.

Activation energy for hardening was estimated to be around 20 kcal/mol which is a similar value obtained in the toughness loss of dual phase stainless steel (8), and probably the same precipitates described in (6,9), formed in the matrix. This fact may suggest that there exists a possibility of grain interior embrittlement in addition to grain boundary embrittlement.

If this alloy would be embrittled by the grain interior hardening, molybdenum content must be controlled below some proper level. Present data are not enough to suggest whether the grain interior embrittlement would occur or not and to assume how much is the proper level of molybdenum content. Further study is required to prove the phenomena with more accuracy, in which material must be aged at lower temperature for longer time.

Effect of heat treatment was also obtained. The increase in hardness in the alloy with heat treatment HT II was slower than that in the alloy with heat treatment HT I (Fig. 8).

Changing the heat treatment may be very effective to control the degradation process of the alloy. As shown in the results, HT II suppressed the loss of toughness, and HT II may make the alloy aging-resistant.

The reason why heat treatment HT II delayed the decrease in toughness is not clear, but from the fact that HT II suppressed both the loss of toughness and hardening, it can be speculated that heat treatment HT II stabilizes both phosphorus and molybdenum by producing some molybdenum-phosphorus compound.

Finding out a good heat treatment for the alloy will be another subject for future study.

SCC test result

SCC susceptibility of the alloys in high temperature water was evaluated by crack depth change as shown in Figs. 11 and 12. After aging for 10,000 hours, SCC susceptibility of all the alloys became slightly higher than that of the unaged condition. As mentioned in the AES observation, phosphorus segregation was a typical phenomenon occurring at the grain boundary after the long time aging, and might have enhanced the IGSCC of the aged alloy. Molybdenum also appeared to enhance the IGSCC, but the effects of molybdenum and the heat treatment as well were very small.

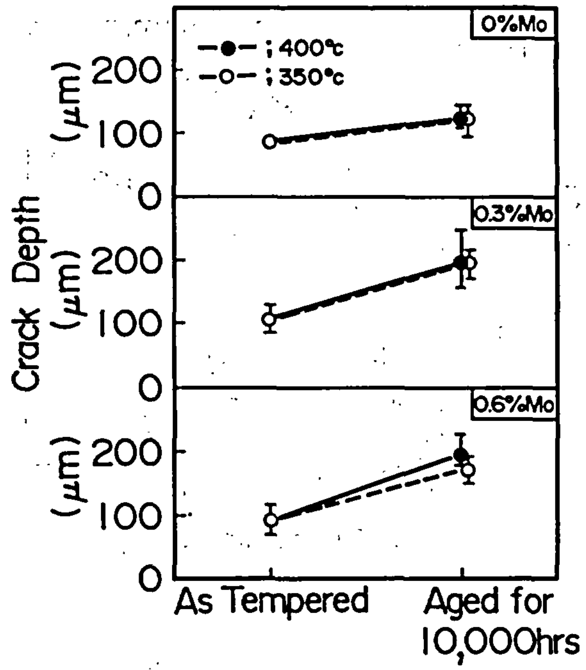


Figure 11. SCC crack depth observed after CBB test in unaged and aged alloys M0, M3 and M6 (1).

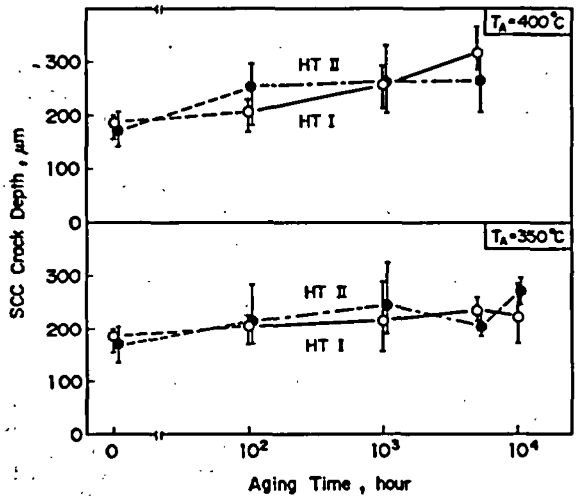


Figure 12. SCC crack depth observed after CBB test in unaged and aged alloys HP.

SCC has occurred intergranularly with the surface pittings and surface oxidation. In this alloy, continuous chromium depleted zone along the grain boundary was not observed in STEM/EDS analysis, even though the isolated nickel-rich phase, in which chromium was depleted, was observed near the grain boundary(1). Local cell reaction between matrix and segregated elements, such as phosphorus and molybdenum, may be a cause for intergranular cracking.

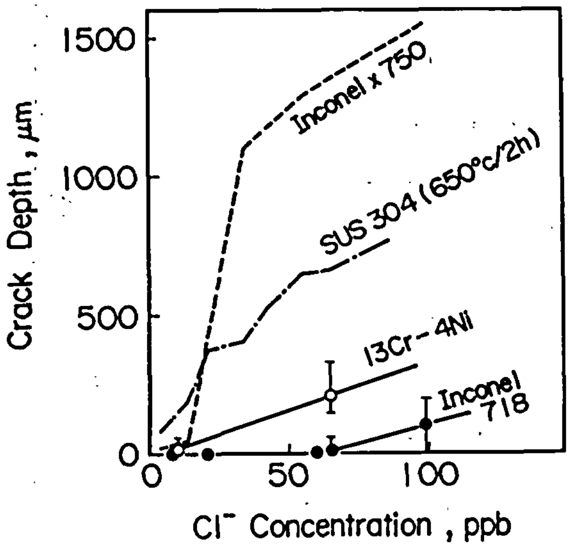


Figure 13. SCC susceptibility comparison of the alloys, as a function of chloride concentration in high temperature water, tested at 288°C by CBB technique. Heat treatments for alloy X-750 (15) and 718 (16) were 1093°C/1h + 704°C/20h and 1010°C/1h + (718°C/8h, F.C. to 621°C)/total 18h, respectively.

Even if the specimens showed susceptibility to IGSCC, 13Cr-4Ni alloy is less susceptible to SCC than other materials used in the light water reactors. As shown in Fig. 13, in which the SCC crack depth by the CBB test is indicated as a function of chloride concentration of test water, SCC susceptibility of 13Cr-4Ni alloy is lower than that of alloy X-750 and sensitised AISI 304 (15,16). Therefore, it can be concluded that 13Cr-4Ni-Mo alloy has SCC susceptibility, but it is not so high as the other materials normally used in the reactors, even after the long time aging at the operation temperature of the light water reactors.

As indicated in this report, many factors, such as material composition, fabrication process including initial heat treatment of the alloy, aging temperature etc., affect the metallurgical change in the aged alloy. Mechanical and SCC properties of the aged alloy are the result of complex effects of those factors. Accordingly, further studies from the material science side will be required, in order to know what would occur in the material used in LWR environment and to perform PLEX.

CONCLUSIONS

Thermal degradation of 13Cr-4Ni-Mo martensitic stainless steel has been investigated and conclusions were obtained as follows:

1. Impact energy of 13Cr-4Ni-Mo alloy decreased by 400 and 350°C aging and the activation energy for the loss of toughness was assumed to be 37-39 kcal/mole.

2. From the AES and TEM analysis, it was conjectured that molybdenum was beneficial for suppressing the phosphorous segregation to the grain boundaries.

3. The alloy was age-hardened by precipitates. Activation energy for hardening was estimated to be around 20 kcal/mol, and molybdenum accelerated the hardening.

4. Heat treatment prior to aging could change the degradation behaviour. HT II (990°C/4h+600°C/4h+580°C/4h) was a better heat treatment than HT I (1050°C/5h+1020°C/1h+640°C/1h+620°C/1h).

5. SCC occurred intergranularly under simulated BWR water condition and was slightly enhanced by the aging. But, compared to the other alloys such as alloy X-750 and sensitized AISI 304, 13Cr-4Ni was less susceptible to SCC, even in the aged condition.

6. Further studies from material science side are required to predict the properties of aged material and to perform PLEX.

ACKNOWLEDGEMENT

The authors would like to send many thanks to Mr. T.Okada of Kensa Engineering Co. for his skillfull help in experimental work, and also to Mr. M.Hishida, T.Okada and H.Sakamoto, nuclear division of Toshiba, for the beneficial discussion on the study and giving the opportunity to perform this study.

REFERENCES

1. M. Tsubota, K. Tajima, H. Sakamoto and T. Okada: The 3rd International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Aug. 30-Sept. 3, 1987, Traverse City, MI.
2. J.J. Heger: Metal Progress, 1951, vol. 8, p.55.
3. Q. Zhe, Y.Q. Weng, S.C. Fu and C.J. Macmahon, Jr: EPRI Repot CS-3248, 1983.
4. S.M. Brummer, L.A. Charlot, M.T. Thomas and R.H. Jones: EPRI Report RD-3859, 1985.
5. Y.Meyzaud and R. Cozar: Proc Top Conf Ferritic Alloys Use Nucl Energy Technol, 1983, p. 27.
6. C.K. Chopra and H.M. Chung: DOE Report NUREG-0975, vol. 3, 1985, p. 284.
7. S.G. Druce, J.M. Titchmarsh, G. Jordan and A. James: UKAEA Report AERE-R-11626, 1985.
8. O.K. Chopra and H.M. Chung: The 3rd International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Aug. 30-Sept. 3, 1987, Traverse City, MI.
9. H.M. Chung and O.K. Chopra: The 3rd International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Aug. 30-Sept. 3, 1987, Traverse City, MI.
10. Ph Dumoulin, M. Guttmann, M. Foucalt, M.Palmier, M. Wayman and M. Biscondi: Metal Sci., 1980, vol. 10, p. 1.
11. M. Guttmann, Ph Dumoulin and H. Wayman: Metall. Trans. A, 1982, vol. 13A, p. 1693.
12. P. Billard, J.R. Donati, D. Guttmann, M. Guttmann, S. Licheon and J.C. Van Duysar: Proc Top Conf Ferritic Alloys Use Nucl Energy Technol, 1983, p. 425.
13. M. Akashi and T. Kawamoto: IHI Eng. Rev., vol. 11, (1978), p. 1.
14. J.F. Copeland and A.J. Giannuzzi: EPRI Report NP-3673-LD, 1984.
15. K. Hattori, M. Tsubota and T. Okada: Corrosion, vol. 42, 1986, p. 531.
16. M. Tsubota, K. Hattori, T. Kaneko and T. Okada: presented at EPRI Work Shop on "Advanced High Strength Materials of LWR," Clearwater Florida, March 1986.

WALL THINNING IN NUCLEAR PIPING STATUS AND ASME SECTION XI ACTIVITIES

Spencer H. Bush and Bindi Chexal

Abstract

Wall thinning due to wet steam is a well known phenomenon in both fossil and nuclear power plants; however, the widely reported wall thinning and failure of a feedwater line in the Surry Power Plant was quite unexpected because it occurred in single-phase liquid water. Since the Surry failure, wall thinning has been reported in other nuclear plants, demonstrating single-phase erosion-corrosion to be a generic issue.

Critical parameters influencing wall thinning in single-phase liquid water are: alloy composition of the pipe; degree of turbulence in a given location; pH level; oxygen content, linear flow velocity; and temperature. Other factors may influence rate of attack and type of attack; e.g., uniform wall thinning, slot-type attack, or highly localized wall thinning.

The USNRC has requested action by the ASME Codes, specifically Section XI. This paper discusses the wall-thinning problem and current programs, particularly at EPRI aimed at its definition and resolution, and the proposed Section XI plan of attack. The Section XI plan is based on the maximum utilization of the available data generated by EPRI and by the USNRC. Section XI plans on covering nondestructive examination (NDE) related to where and how often to examine and with what tools; e.g., ultrasonics or radiography. In addition, the issue of wall-thinning evaluation analogous to flaw evaluation will be covered. It is hoped that there will be reduction in level of examination to partially compensate for the added requirements of NDE to detect wall thinning.

Introduction

The failure of a feedwater line at Surry-2 on December 9, 1986 surprised both the utility industry and the USNRC. A steam valve closure increased the pressure about 20 percent, which was sufficient to fail a 90° elbow in the balance-of-plant (BOP) portion of the 18-inch feedwater line. A section of pipe about 2 feet by 3 feet blew out resulting in complete separation. The reaction forces moved the pipe about six feet. Four of eight contractor employees who were in the area subsequently died of severe burns.

While the failure occurred in the non-safety BOP, activation of the fire protection system led to shorting of electrical circuits and seepage of carbon dioxide and other chemicals into the control room.

An examination of the failed section revealed thinning to below a tenth of an inch. The cause was determined to be single-phase erosion-corrosion. Significant parameters affecting erosion-corrosion are chemical composition of the pressure boundary material; pH level, temperature and oxygen content of the coolant; and coolant flow linear velocity and turbulence.

Subsequently, extensive erosion-corrosion was reported at the Trojan plant in the secondary piping inside containment. A reexamination of previous failures and review of new inspection data gathered worldwide revealed that single-phase erosion-corrosion was more prevalent than had been recognized. In fact it represents a generic problem. Table 1, taken from Table 3.1 of Reference 1, a General Accounting Office (GAO) report on erosion-corrosion, confirms that several nuclear plants have suffered single-phase erosion-corrosion.

Clearly, wall thinning due to single-phase erosion-corrosion is a time-dependent phenomenon. Under extreme conditions with plain carbon steel piping, severe attack may occur in a relatively short time. With more benign conditions and/or higher alloy levels in the pressure boundary material, the attack may be delayed many years or may not occur during the plant life.

The remainder of this report will deal with two aspects of the wall-thinning problem:

1. What causes single-phase erosion-corrosion, what factors cause or mitigate such attack, and where is such attack expected to occur.
2. What corrective actions are planned, together with a discussion of the current status of these corrective actions within Section XI of the Boiler and Pressure Vessel Code.

Erosion-Corrosion, its Causes and the Prediction of Occurrence

Erosion-corrosion or, as it is more accurately called, flow assisted corrosion is the process by which carbon or low alloy steel components lose material through the dissolution of the poorly adherent magnetite layer. This phenomenon normally occurs in flowing, deoxygenated water with a pH between 7.0 and 9.5. It is not a classical erosion process in that the material loss is not caused by a mechanical process. Therefore, there is not a threshold velocity below which there is no material loss, and above which there is extensive damage.

Single-phase flow erosion-corrosion is most likely to occur in by-pass lines such as recirculation flow lines around pumps and control valves, downstream of flow control valves (angle valves in particular) and in elbows in close proximity of other fittings. Instances of single-phase erosion-corrosion have been reported in other fittings such as at the small diameter end of a diffuser, J-tubes in steam generators, etc.

Erosion-corrosion is caused by a complicated interplay of a number of parameters. A large body of experimental work has identified several key variables that influence the rate of attack. These variables are listed below with an indication of how they impact the material loss behavior.

Variable	Erosion-Corrosion Increases If Variable Is
Fluid velocity	Higher
Fluid pH level	Lower
Fluid oxygen content	Lower
Fluid temperature	250-400°F

Component geometry	Such as to create more turbulence
Component chromium content	Lower
Component copper content	Lower
Component molybdenum content	Lower

The complexity of these variables and their interrelation are such that a mathematical model which considers all of the variables is required to make erosion-corrosion predictions with any accuracy. This predictive capability helps avoid wholesale, random and non-productive inspection efforts.

The CHEC™ (Chexal-Horowitz-Erosion-Corrosion) computer program[2] was developed by EPRI to meet this industry need. The general formulation of the model used in CHEC is a series of factors which, when multiplied together, yield the predicted erosion-corrosion rate. Since some of the factors are interrelated, the model is not linear. The formulation is as follows:

$$E = F_1(T) \cdot F_2(AC) \cdot F_3(MT) \cdot F_4(O_2) \cdot F_5(pH) \cdot F_6(G)$$

where E = erosion rate

$F_1(T)$ = factor for temperature effect

$F_2(AC)$ = factor for alloy content effect (chrominum, copper and molybdenum content)

$F_3(MT)$ = factor for mass transfer effect (flow rate, piping diameter)

$F_4(O_2)$ = factor for oxygen effect

$F_5(pH)$ = factor for pH effect (amine type)

$F_6(G)$ = factor for geometry effect

Since the interrelation between these parameters was not initially apparent, the formulation was developed empirically.

A large data base was assembled from various laboratories, and an optimum model was obtained using an iterative procedure. This model followed all of the experimental trends, and correlated well with the bulk of the laboratory data. The model was further refined by comparing the predictions of the model with wall thickness inspection data obtained from several nuclear and fossil power plants, and with further laboratory research (particularly to take into account various geometrical mass transfer enhancement factors).

The correlation between CHEC predictions and the plant inspection data has been very good. The ability of the code to predict single phase erosion-corrosion rates within a ±50% band, given accurate input data, has been demonstrated. This agreement is much better than other known erosion-corrosion correlations, and CHEC continually is being made more accurate by data and analyses feedback from nuclear and fossil plant users and by the results of continuing research.

An analysis with CHEC involves two passes. In the initial pass, all piping components included in the analysis are ranked in order of their relative susceptibility to erosion-corrosion, as well as in order of remaining lifetime. This output can be used to identify locations for an initial plant inspection.

When wall thickness inspection data from a minimum of ten components become available from this initial inspection and are input for the second analysis pass, a plant specific correction is made to the predictive model. This correction accounts for uncertainties in the plant data, particularly in the area of chemistry history, and corrects systematic discrepancies caused by plant operation. Thus, if the pH is not known accurately for the entire plant history, the program will correct, within limits, for this uncertainty.

Results of the second pass include quantified predictions of erosion-corrosion rates and the times until minimum code allowable wall thicknesses are reached for each piping component included in the analysis. This output can be used to help establish inspection sample expansion, if required by initial results, and for planning the locations and timing for future inspections.

Current Mitigation Options

Depending upon the extent of wall thinning, utilities are applying several options to rectify the problem. These include:

- o implementing water chemistry changes,
- o changing piping design/layout to improve flow geometries,
- o repairing or replacing with more resistant materials.

Water chemistry changes are attractive in that they offer a means of prolonging the life of existing piping. Two water chemistry variables, other than temperature, have been shown to have strong effects on the rate of erosion-corrosion, namely pH level and dissolved oxygen content.

In PWRs, increasing the pH level can significantly reduce the rate of erosion-corrosion. In systems containing only ferrous alloys, pH control in the 9.3 to 9.6 range has been shown to yield acceptably low values of iron release from typical carbon, low alloy, and stainless steels in power systems. In plants with copper alloys in the feedwater heaters or condenser, a lower range (8.8-9.2) is mandated as copper released from some alloys has shown to increase markedly when the pH is above 9.2. The PWR secondary water chemistry guidelines do allow for operation above pH 9.2 if individual plant experience shows that copper transport does not increase significantly. The adoption of morpholine rather than ammonia as the pH control additive has also been used to reduce the rate of erosion-corroson.

BWR water chemistry differs significantly from that in a PWR. First, chemical additives are not employed routinely. Second, significant oxygen levels exist in the condensate, feedwater, and steam trains. A data base exists illustrating the beneficial effect of maintaining the oxygen concentration in BWR feedwater and condensate above the minimum value given in industry guidelines. Although decreases in the release of iron from ferrous materials would not be considered significant with respect to reduction of deposits on fuel or primary systems activity levels, operation near the 50 ppb upper limit of the indicated achievable range could reduce the probability of flow assisted corrosion in single-phase regions.

Replacement of carbon steel piping components with low alloy or low carbon grade austenitic stainless steel materials has also been used successfully to mitigate erosion-corrosion. Current data suggest that an alloy containing 1/2 to 1 percent chromium would provide adequate resistance in single-phase systems. Two low alloy steels which are available in a variety of sizes and for which there has been considerable power plant experience are 1-1/4 Cr-1/2 M_o (P11 grade) and 2-1/4 Cr-1 M_o(P22 grade). As these low alloy steels have almost the same mechanical properties at the operating temperatures of interest, replacement piping of this material can be installed with the same geometry and unit weight as the original carbon steel components. Additionally, the thermal stresses and nozzle loadings are of little consequence due to the similarities in the coefficients of thermal expansion. As a result, the substitution of either of these grades is generally straightforward, and any design analysis should be minimal for the same configuration. One disadvantage is that both the P11 and P22 Grades require special considerations for welding, especially preheat and postweld heat treatment. However, these considerations are well documented and represent standard practices in the industry.

Austenitic steels also have excellent resistance to erosion-corrosion. Low carbon grades are preferable because of better intergranular stress corrosion cracking (IGSCC) resistance. The candidate materials are 304L, 316L, and 347L. These materials are readily available and do not require preheat or postweld heat treatment. The disadvantages of austenitic stainless steels are that piping reanalysis is required due to a higher thermal expansion coefficient (1.4 X carbon steel); the bimetallic welds need special attention; and susceptibility to chloride stress corrosion raises concern over the chloride contaminants in thermal insulation.

An alternative to component replacement is an internal coating procedure whereby an erosion-corrosion resistant material is applied as a thin (~20 mil) overlay to the inside surface of carbon steel piping. Two potential procedures have been used successfully in Europe for several years. Both involve flame spraying multiple layers onto the carbon steel surface. The first layer is of an alloy content to provide sufficient adhesion to the carbon steel, with the last layer the protective coating. One procedure uses a stainless steel protective coating, while the other uses a high nickel alloy. The coatings can be applied economically in the field without removing the piping, but such application would be limited to sizes large enough for worker access (~24 inch diameter and above). The coating can be applied to smaller piping in the shop, but requires cutting out existing piping and replacing with coated sections and is significantly more expensive. Both of these procedures will soon be made commercially available in the U.S.

Where wall thinning is local, weld repair is an attractive temporary fix. An appropriate weld alloy can be deposited to fill an eroded cavity to bring the pipe back to its original thickness. However, firsthand inspection of internal weld repair areas that have been placed back in service has shown this not to be a lasting repair. It has been observed that in many instances the areas adjacent to the weld repair were even more vigorously attacked and degraded, while at the same time leaving the welding or cladding itself almost untouched. The area adjacent to the weld has increased susceptibility of erosion-corrosion in the heat affected zone (HAZ). Changes in material grain structure alter the physical characteristics, which allows a more vigorous attack in these regions. However, experience in Europe has shown that applying a flame sprayed protective coating to the repaired piping (covering the area beyond the HAZ) can prevent such attack and make the repair more permanent.

A History of ASME Activities Leading to Current Corrective Actions

Shortly after the Surry-2 failure a letter from Robert Bosnak of USNRC was sent to John Fernandes, a Senior Vice President of ASME and the Chairman of the Council on Codes and Standards (CCS). This letter requested appropriate action by ASME. Recommendation 4 from the review of the Surry-2 failure states in part:

> "The American Society of Mechanical Engineers (ASME) should consider the need for providing appropriate guidance to system designers on the subject of erosion and erosion-corrosion in its conventional pressure piping and nuclear piping Codes and Standards. Additionally, the Subcommittee on Nuclear Inservice Inspection (SC XI) of the ASME Boiler and Pressure Vessel Committee and the ASME groups active in plant aging and life extension matters should be made aware of the need to consider requiring pipe wall thickness measurements in their respective programs."

The first portion of the above recommendation deals with B31.1 and Section III and will not be addressed here. Only the Inservice Inspection aspects will be considered in detail; however, Section XI has the lead role within ASME on matters pertaining to plant life extension, and members of Section XI have been kept aware of the continuing status of the wall-thinning problem.

A letter from John Fernandes requested appropriate action of the appropriate ASME groups by December 1987. Section XI responded to this request in a letter to him in October 1987. The Section XI approach was presented to CCS in December, after which they instructed Section XI to proceed along the lines presented in the letter. Basically, the Section XI position was to leave the BOP as a utility responsibility and concentrate on safety-related systems. Nominally these are expected to be Classes 1 and 2 for BWRs and Classes 2 and 3 for PWRs.

A Working Group on Pipe Wall Thinning has been established covering the areas of nondestructive examination (NDE), flaw/wall thinning and analytic evaluation techniques, and criteria for the selection of areas to be examined as well as the specifics on how much to examine. The intent has been to take maximum advantage of the work of others, particularly EPRI, so that a Code revision can be made in a relatively short time period. We expect to take advantage of the output of the EPRI CHEC computer program to optimize selection of regions considered to be most susceptible to single-phase erosion-corrosion. The bases for this selection were developed in the previous section of this paper.

With regard to NDE, the situation is not clear because at least three distinct forms of erosion-corrosion attack have been observed. These are:

1. uniform wall thinning,

2. highly localized wall thinning, often in widely separated regions (the area of such thinning may be 20 to 50 square inches), and

3. severe and highly localized axial slot attack analogous to a sharp slit generated by a cutting tool.

The differences in attack pose problems to NDE on where and how to look. If radiography is used, more complete coverage is possible but it poses problems to workers in the vicinity. Ultrasonics, usually a zero degree beam, should detect thinning, providing the correct regions and specific locations are examined.

Once wall-thinning of types (1), (2), or (3) is detected a decision must be made on timing of replacement/repair, or continued operation. Fracture mechanics procedures have been developed for relatively thin-walled sections typical of most piping. Hopefully, these procedures will permit a less conservative approach to replacement than the one used to date of replacing pipe if it is below Code minimum wall.

Our optimistic goal is to complete the activities of the Working Group on Pipe Wall Thinning no later than early 1989 and report positions in Code language to the appropriate Working and Subgroups of Section XI with the goal of Subcommittee action in 1989.

The most difficult problem will be putting in Code language the locations to be examined, since they will vary from plant to plant. Basically, all ASME Codes are deterministic so it is relatively straightforward to define a weldment for NDE. Unfortunately, there are several problems that do not lend themselves to a deterministic approach such as:

1. what specific components should be examined and how often,

2. what NDE technique(s) should be used, and

3. how large a grid pattern needs to be applied to components for NDE.

An expansion of item (3) indicates the dilemma we face. One could apply a four-inch grid, which has been done by several utilities. However, some use a UT transducer at the nodal points only (<1% of the area), others use a serpentine pattern along the sides of each grid (10-40%), while a few utilities examine the area within each grid (~100%). Another aspect is that a four-inch grid could miss slot type attack; this means we will be plowing new ground in the Code in attempting to optimize examination without applying excessive requirements.

Hopefully, all of the problems will be resolved and a consistent approach developed for systems of concern, primarily the heater drain, feedwater and condensate trains.

References

1. "Action Needed to Ensure That Utilities Monitor and Repair Pipe Damage," Report to the Honorable Edward J. Markey, House of Representatives, United General Accounting Office, GAO/RCED-88-73, March 1988.

2. V.K. Chexal, E.B. Dietrich, J.S. Horowitz, G.A. Randall, V.C. Shevde, and J.A. Thomas. "CHEC (Chexal-Horowitz-Erosion-Corrosion) Computer Program User's Manual," NSAC-112, Electric Power Research Institute, Palo Alto, CA, Final Report, February 1988.

TABLE 1

NUCLEAR PLANTS REPORTING EVIDENCE OF EROSION/CORROSION

Plant	State	Date of Initial Reactor Operation	Location Where Erosion/Corrosion Was Detected				
			Single-Phase	Elbows	Straight Pipe	Fittings	Other
San Onofre Unit 1	California	June 1967					X
Haddam Neck	Connecticut	July 1967	X				X
Oyster Creek	New Jersey	May 1969	X	X			
Dresden Unit 2	Illinois	January 1970	X	X			
H B Robinson Unit 2	South Carolina	September 1970					X
Pilgrim Unit 1	Massachusetts	June 1972	X	X			
Surry Unit 1	Virginia	July 1972	X			X	
Turkey Point Unit 3	Florida	October 1972				X	
Surry Unit 2	Virginia	March 1973	X			X	
Fort Calhoun	Nebraska	August 1973	X	X	X		
Fort St Vrain	Colorado	January 1974			X		
Duane Arnold	Iowa	March 1974		X	X		X
Arkansas Unit 1	Arkansas	August 1974	X	X			X
Rancho Seco	California	September 1974			X		
Calvert Cliffs Unit 1	Maryland	October 1974		X	X		X
Millstone Unit 2	Connecticut	October 1975	X	X			X
Trojan	Oregon	December 1975	X	X	X		X
Calvert Cliffs Unit 2	Maryland	November 1976		X	X		X
Salem Unit 1	New Jersey	December 1976	X				X
D C Cook Unit 2	Michigan	March 1978		X			
North Anna Unit 1	Virginia	April 1978	X	X	X		
Arkansas Unit 2	Arkansas	December 1978					X
North Anna Unit 2	Virginia	June 1980		X	X		
Sequoyah Unit 1	Tennessee	July 1980	X	X	X		
Salem Unit 2	New Jersey	August 1980	X				X
Sequoyah Unit 2	Tennessee	November 1981	X	X			
San Onofre Unit 2	California	July 1982	X				X
San Onofre Unit 3	California	August 1983	X				X
Diablo Canyon Unit 1	California	April 1984		X	X		
Callaway	Missouri	October 1984	X	X			
Diablo Canyon Unit 2	California	August 1985		X			X
River Bend Unit 1	Louisiana	October 1985					X
Perry	Ohio	June 1986			X		
Shearon Harris	North Carolina	October 1986					X
Total			18	18	12	3	17

AGING AND REACTOR WATER EFFECTS ON FATIGUE LIFE

By

W. J. O'Donnell
J. S. Porowski
E. J. Hampton
M. L. Badlani
G. H. Weidenhamer
D. P. Jones
J. S. Abel
Brian Tomkins

ABSTRACT

Methods of including aging effects and reactor water enhanced crack propagation rates in Codified S-N fatigue life assessment curves are presented and illustrated. Such methods are essential because it is not feasible to produce experimentally based S-N life evaluation curves for all of the reactor materials of interest under all of the relevant cyclic rate and environmental conditions of interest within available finite research funding.

Reactor water environmental effects are known to accelerate fatigue crack growth rates in reactor pressure vessel and piping materials. Recently developed advanced elastic-plastic fracture mechanics technology [Ref. 1] is used herein as a means of correcting S-N fatigue life evaluation curves for measured environmental crack growth rate effects. As an important illustration, ASME Code Section XI reactor water crack growth rate curves are used to generate revised new Section III and VIII fatigue design curves for A106 reactor piping. Reactor water effects on the fatigue life are found to be quite significant, and their inclusion in the S-N curves greatly improves the technical basis for assessing the residual component life which meets ASME Code safety margins for cumulative fatigue.

INTRODUCTION

Recognizing the increasing importance of nuclear plant license renewal, NRC published a Program Plan for Nuclear Plant Aging Research in July of 1985 [Ref. 2]. The research described in this Plan is intended to resolve issues related to the aging and service wear of equipment and systems at commercial reactor facilities and their possible impact on plant safety. Considerable emphasis is on the mechanisms of material and component degradation during service. One of the major program goals is to identify and characterize aging and service wear effects which, if unchecked, could cause degradation of structures, components, and systems and thereby impair plant safety.

Nuclear utilities need a sound technical basis for making plant, system and component repair and replacement decisions [Ref. 3]. Since a new plant requires many years to plan and construct, decisions concerning the expected operating life of existing plants must be made long before the end of the 40 year licensed life of the plant. The reliability and dependability of operating plant components are very important to utilities operating nuclear power plants. Forced outages are disruptive and costly.

In spite of recent advances, inspection technology is not yet totally reliable, and there are areas not accessible for inspection. S-N technology provides a reliable means of evaluating the safe operating fatigue life without depending on inspection results. It also covers vibratory and thermal high cycle loading conditions which could result in the rapid propagation of cracks too small to be detected by in-service inspection.

This report focuses on the development of evaluation methods which can be used to accurately predict the reliability and structural integrity of components, systems and piping. These criteria are intended to assure that regulatory and Code[1] safety margins used to design and license the plant are maintained during the extended period of operation, and that the reliability and dependability of the hardware is not seriously compromised by corrosion-assisted fatigue.

The fatigue design evaluation criteria for nuclear components in Section III of the ASME Code [Ref. 99] are based on 30-year-old technology and do not account for important environmental and aging factors which must be considered in evaluating the service life of components and piping. On-going systematic long-term research by NRC and others is based on fracture mechanics technology for the initiation, propagation and instability of cracks. The analytical methods developed in Ref. [1] combine the existing Codified S-N technology with elastic-plastic crack propagation technology to include reactor water environmental effects on crack growth rates in improved S-N criteria for nuclear components. These methods can also be extended to account for imperfections and residual stresses in weldments.

Most LWR plants have not experienced all the hypothetical transients and earthquakes included in the original fatigue design analyses. Thus, these plants will not have experienced as much fatigue usage or damage as anticipated in the original design stage, and there should be margin for life extension beyond the original 40-year life. However, the results of environmental degradation testing during the past fifteen years have shown

---

[1] "Code" in this report refers to the ASME Boiler and Pressure Vessel Code.

that such effects are more deleterious than anticipated when the ASME Code adopted the current S-N fatigue design curves. Therefore, environmental and aging effects must be taken into account when reevaluating safe operating life. Improved fatigue life evaluation curves for A106 piping steel which include reactor water crack growth rates from Section XI of the ASME Code are obtained herein using the methods of Ref. [1].

Extensive data analyses are needed to evaluate the effect of reactor water environments on the crack initiation phase of fatigue failure and to evaluate the effects of weldment imperfections and residual stresses. The resulting new S-N curves will provide a means for utilities to evaluate current fatigue damage in their plants, taking into account the cumulative damage from operating transients and cycles which the plant has experienced. The safe residual life can then be evaluated using the same curves in order to assure compliance with Code safety margins. This plant life extension approach is applicable even where in-service inspections are not feasible. Of course, it does not account for fabrication or stress corrosion-induced cracking, which can be found by in-service inspection and evaluated using Section XI criteria.

Considerable fatigue data on BWR environmental cracking effects was generated in an EPRI-sponsored program [Refs. 4 and 5]. Much of this data was obtained on notched specimens and cannot be generalized to other geometries. The elastic-plastic crack analysis approach herein provides a method of using that data to evaluate environmental effects on crack initiation and propagation. Such results can then be used to generate S-N fatigue curves.

Research carried out under NRC sponsorship [Refs. 6 - 12] provides crack growth data directly usable in the general method of Ref. [1] developed herein for including such results in S-N curves. The effects of various material, environmental and operating parameters on crack initiation and propagation are being evaluated in on-going research worldwide (see for example Refs. [5 - 45]). Thus, these results can now be used to improve the S-N fatigue life evaluation curves.

COMPONENT FATIGUE TEST EXPERIENCE

Numerous fatigue tests have been run on pressure vessels, piping and components following the famous early work on piping components by Markl [Refs. 46 thru 49] beginning in the 1940's. The Pressure Vessel Research Committee and Atomic Energy Commission sponsored early low cycle fatigue tests on full size pressure vessels incorporating a variety of nozzle configurations of interest to the reactor designer and pressure vessel industry at large [Refs. 50 thru 53].

The local stress conditions were carefully measured using brittle coatings and nearly 900 strain gages. Peak stresses occurred at the inside corners of the nozzles and fatigue cracks developed at these locations. Crack initiation and crack growth were observed. A comparison of the crack initiation and failure points with the ASME Code design curves provides an indication of the safety margins provided by the current design criteria. No crack progressed thru-the-wall in less than three times the allowable number of design cycles.

While this comparison indicates that the current ASME Code design curves do provide the intended safety margins (see also Ref. [53]), the tests did not include the effects of reactor water corrosion-assisted fatigue crack growth anticipated during the operation of Light Water Reactors. Laboratory test results indicate that crack growth rates of LWR ferritic steels in reactor water at 550°F may be an order of magnitude higher than in air for certain material and water chemistry conditions with ΔK in the intermediate range. Moreover, the tests did not include the spectrum of thermal and vibration cycles experienced under actual reactor operating conditions.

Since these tests were run, numerous such tests have been run on various components worldwide. In the U. K., where elastic shakedown is used as a basic design criteria, extensive tests have been run on full-size components.

Cracks often initiate early at acceptable imperfections in weldments. While some of these cracks stop propagating as they progress beyond the stress raising influence of the imperfection, others may continue to propagate, limiting the fatigue life. In the high-cycle regime, cracks initiate late in the fatigue life, and propagate thru-wall in a short time.

The NRC sponsored a Battelle study [Ref. 54] and evaluation of available fatigue test data on piping products which was intended to determine the margin of safety of the ASME Code design curves. These tests of piping products form important bases and justification for the fatigue evaluation procedures used for nuclear power plants. However, as pointed out in the report, the tests were run over a relatively short period of time (days or weeks) as compared to the 40-year design life of a nuclear plant. Accordingly, the tests do not encompass environmental effects. Corrosion-assisted fatigue crack growth is chemistry, temperature and cyclic rate dependent. NUREG/CR-0325 [Ref. 54] concludes: "The most important discrepancy exists in assessing the effect of the environment over the 40 years of PWR plant life. If the environmental attack is significant, the cold leg behavior may not be conservative with respect to the fatigue test results. To accurately evaluate the margin of safety inherent in the cold leg piping, an analysis technique that accounts for the environmental factors must be employed."

As part of the Nuclear Plant Aging Research Program, the NRC sponsored an ORNL study [Refs. 55 - 58] of available sources of LWR operating experience contained in Licensee Event Reports (LER's) to identify and evaluate age-related events and trends which could result in compromising a safety function. Fatigue, corrosion, vibration and cracks were found to be very significant factors. Data collected [Ref. 55] on domestic commercial nuclear power plants covering 1969 to 1983 yielded 324 failures specifically reported as fatigue failures. 75 percent of these failures were judged to be degraded failures which placed plant operation outside the Technical Specifications, and 25 percent were judged as catastrophic failures. An additional 259 failures were keyworded "crack," 165 were reported as "vibration" failures, 110 as "stress-corrosion" failures and 414 as "corrosion" failures. These results indicate that corrosion effects contributed to a large percentage of the failures, and that corrosion-assisted fatigue should be included in the S-N design curves.

Typically, fabrication-related problems tend to show up early in the operating life of a plant. This is followed by a long period of relatively trouble-free operation until fatigue and other aging phenomena begin to produce failures. Such failures then tend to occur with increasing frequency as the systems, components and piping approach the end of their useful lives.

AGING EFFECTS ON FATIGUE LIFE

The effects of aging on fatigue life have yet to be quantified and methods of doing so are described herein. Concerns include thermal aging and the associated loss of toughness, reduced ductility, via increased notch sensitivity.

Thermal embrittlement, well known in cast stainless steels, is expected to be far less significant in wrought materials. The associated loss in toughness and ductility would not be expected to have a significant effect on fatigue life, except in cast stainless steels and possibly in weldments.

Fatigue data correlates very consistently with fracture ductility, following the Langer-Coffin relation from a tensile test to the high cycle regime:

$$S_a = \frac{E}{4\sqrt{N}} \ln\left[\frac{100}{100 - R.A.}\right] + S_e$$

where

E = elastic modulus (psi)

N = number of cycles-to-failure

S_e = endurance limit (psi)

R.A. = Reduction in Area in tensile test

This equation was originally derived by using the true strain at fracture in a tensile test as a point on the S_a vs. N fatigue failure curve, assuming the existence of a well-defined endurance limit. It provides a good fit of strain-controlled fatigue data for most materials out to 10^6 cycles. Of course, more complex equations with additional parameters can be used to achieve better statistical fits to the S_a vs. N data. Beyond 10^6 cycles, where fatigue is controlled by crack initiation, the effects of loading sequence, mean stress effects, environmental and cyclic rate effects introduce major complexities, which are important for thermal mixing and vibratory loads.

However, the above equation is the basis of the existing ASME Code fatigue design curves and can be used to include any reduction in ductility due to aging in the S-N fatigue life evaluation curves. This is illustrated in Fig. (1), which shows how the

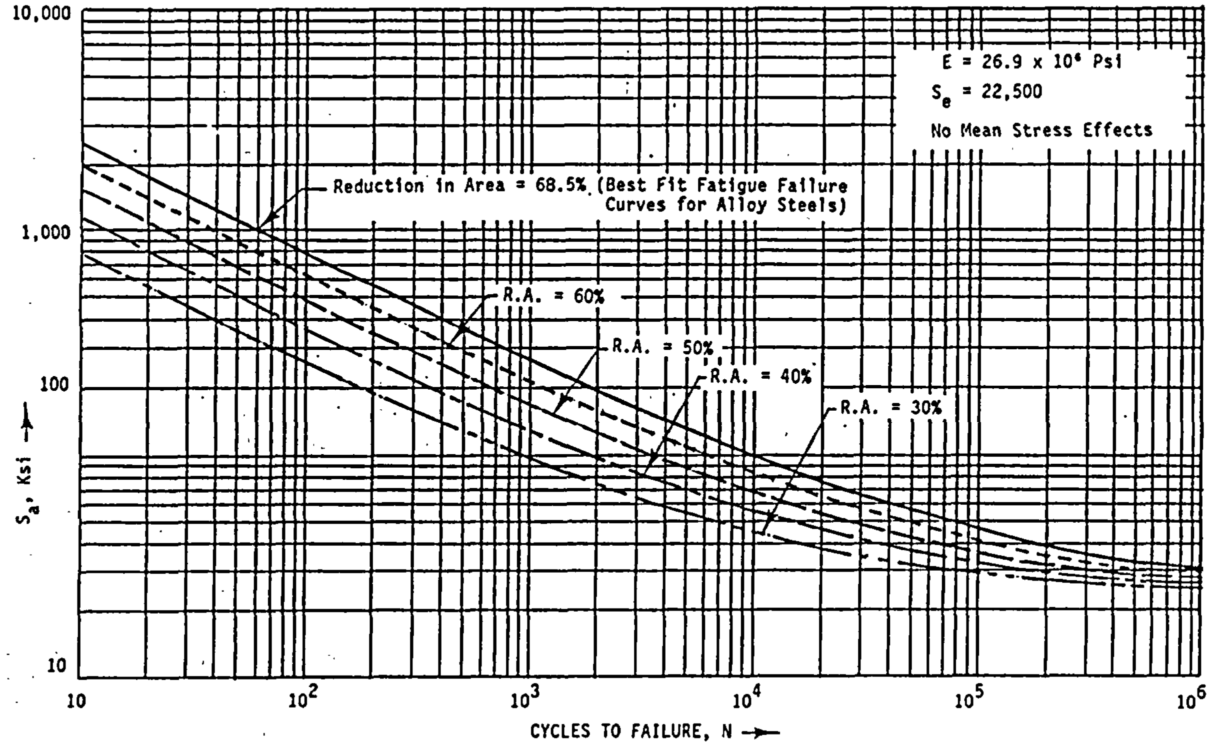


Fig. 1 Illustrates Effects of Hypothetical Reductions in Ductility Due to Aging

failure curves for low-alloy steels would be affected by hypothetical reductions in ductility from Reduction in Area = 68.5% to 60%, 50%, 40%, and 30%, respectively. Losses in ductility of this magnitude due to aging are not anticipated at temperatures below the creep regime.

Concerns have been raised that the crack initiation resistance of weldments could conceivably be reduced by aging. Such effects would reduce the fatigue evaluation curves in the high cycle regime, but would not affect the low cycle regime where crack propagation constitutes more than 90 percent of the failure life. Of course, the methods developed herein to correct fatigue life evaluation curves by considering failure as the sum of the initiation and propagation phases can be used directly to include any change in initiation resistance in the total fatigue life curves.

Increased notch sensitivity has also been cited as a potential aging effect. Fatigue notch sensitivity has been found to be a monotonically increasing function of ultimate strength. This may be largely due to the correlation between strength and ductility, wherein higher strength generally correlates with lower ductility. In fatigue design evaluation practice, fatigue strength reduction factors, K_f, are usually taken to be equal to the local elastic stress concentration factor, K_t. The latter is actually an upper bound for the fatigue strength reduction factor, applicable for very notch sensitive materials, and/or strain raisers having large enough dimensions to grow a crack which is of sufficient size to keep growing beyond the "shadow" of the notch. Most steels used in reactor pressure boundary components are not very notch sensitive. The common practice of using conservatively large fatigue strength reduction factors in fatigue design life evaluations is usually sufficient to cover increases in notch sensitivity with aging.

CYCLIC PLASTICITY IN NUCLEAR COMPONENTS

The process of material fatigue and fracture is one of damage accumulation which can be both local and global. The complexity of the geometric and loading conditions of importance in nuclear components and piping requires that fracture processes be treated within a general and consistent approach. This approach must include consideration of the crack initiation phase as well as both short and long crack propagation. The effects of general plasticity must be included because nominal secondary stresses are permitted to exceed yield and typically do exceed yield in nuclear components and piping.

Linearized secondary stresses in Class (1) nuclear components are limited to $3S_m$ by Section III of the ASME Boiler and Pressure Vessel Code. S_m is the lesser of 2/3 of yield or 1/3 of[2] the ultimate, so that $3S_m$ is about twice yield. Nominal (linearized) thru-the-wall thermal gradient stresses and local stresses which include stress concentration factors are limited only by fatigue, and can be several times yield. Therefore it is essential that any fatigue or fracture analysis methods used to evaluate the safe useful life of nuclear components and piping must be capable of including the effects of plasticity beyond that which can be assessed according to linear elastic fracture mechanics (LEFM) with minor corrections for local plastic zones. The S-N fatigue life evaluation approach with the simplified elastic-plastic strain concentration factors, K_e, in Section III of the ASME Code, includes the effects of such plastic cycling.

LOW-CYCLE FATIGUE CRACK GROWTH

Existing ASME Code Section XI [Ref. 98] flaw evaluation methods are limited to LEFM ΔK concepts, with local plastic zone corrections which are quite small. Elastic-plastic ΔJ crack propagation technology has made great strides in recent years in both its theoretical and experimental facets.

In order to include both elastic and plastic conditions, Rice [Refs. 59 - 61], Hutchinson [Ref. 62] and others analyzed the energy available to drive the crack per unit crack extension. For linear elastic behavior, J is equal to the energy release rate per unit area of crack extension, G. For nonlinear elastic conditions, J is the potential energy difference between two identically loaded bodies possessing slightly different crack lengths. Thus, for either linear or nonlinear elastic behavior, J is the energy at the crack tip per unit area of crack extension, or the crack driving force. With irreversible plastic straining, J is no longer equal to the energy available for crack extension. However, by defining J in the same manner for nonlinear elastic and elastic-plastic conditions, J remains a measure of the intensity of the entire elastic-plastic stress-strain field surrounding the crack tip. In the context of the deformation theory of plasticity, the J-integral denotes the energy released by a unit increase in crack area.

Begley and Landes [Refs. 63 and 64] did considerable early work developing the J-integral as an analytical tool for elastic-plastic cracks using its compliance characteristic. The J-integral has also proven to be of value as a means of evaluating the stress intensity from finite element analyses, using the σ and ε values at locations other than the crack tip. Rice [Refs. 59, 60, 65 and 66] solved two-dimensional crack problems with plastic deformations using a J-integral integration around the tip of the crack. For the power law stress-plastic strain relation, Hutchinson [Ref. 62], and Rice and Resengren [Ref. 67] showed that crack-tip stress and strain singularities are functions of J (see Ref. [68]).

Crack Tip Opening Displacement (CTOD), C*, concepts have also been applied and can be shown to be equivalent to J [Ref. 69] for certain cases. Sehitoglu and Morrow [Ref. 70] have used the cyclic range of the CTOD to characterize thermal fatigue crack growth for AISI 1070 carbon steel.

ΔJ may be meaningfully defined [Refs. 71 through 74] for cyclic loading, where all loads, stresses, displacements and strains in the line integral definition of J are replaced by the

---

[2] In austenitic stainless steels, S_m can be as high as 90 percent of yield where distortions are not critical, so that $3S_m$ can be 2.7 times yield.

amplitudes of the corresponding cyclically varying quantities, $\sigma_a = \Delta\sigma/2$, $\varepsilon_a = \Delta\varepsilon/2$, etc. There is no theoretical objection to a ΔJ so-defined for the materials of interest which behave according to kinematic hardening. In such materials, the hysteresis loops have the same shape as the cyclic stress-strain curves, enlarged by a scale factor of two. Values of ΔJ are computed by treating the cyclic stress-strain curve of the material as a monotonic stress-strain curve.

Goldman and Hutchinson [Ref. 75] have subsequently shown that certain solutions for a simple power hardening relation between the stresses and strains are valid for the incremental theory of plasticity as well as the deformation theory. They also show that the amplitude of the stress and strain of the dominant crack-tip singularity are proportional to J and that simple relationships exist between the J-integral, the applied load, the load point displacement, and the crack opening displacement.

The validity of ΔJ correlations for the elastic-plastic growth of cracks is examined herein based on data for the entire range of loading from purely elastic to the grossly-plastic conditions which occur in low-cycle fatigue testing. Data on A533 pressure vessel steel is used for this purpose because data on similar heats of this material has been reported in the literature for a wide range of sizes, geometries and loading conditions.

Extensive linear elastic fatigue crack growth rate data are given by Paris et al., [Ref.71] for A533B steel. Dowling and Begley [Ref. 72] performed fatigue crack growth rate tests on compact tension fracture mechanics specimens one-inch thick subjected to gross plastic deformation. The data of [Refs. 71 and 72] confirm that the materials used in these tests had essentially the same nominal properties and ambient environment fatigue crack growth rate properties as the material tested by Paris, et al. The specimens tested were identical to the ASTM standard compact tension specimens (ASTM Method E399-73), except for modifications to accommodate clip gages.

The tests were conducted on a closed-loop electrohydraulic testing system. Crack lengths were monitored visually with a low-power traveling microscope. Difficulties with sample load or deflection control for plastic cycling were overcome by controlling the deflection to a sloping line on a load-versus-deflection plot automatically by means of an analog control circuit. In addition, ordinary linear elastic, constant load amplitude, fatigue crack growth rate tests were conducted by Paris, et al., [Ref. 71]. Inelastic values of cyclic ΔJ were determined from the areas under load-versus-deflection curves following Rice, et al., [Refs. 73 and 74] as illustrated in Fig. (2). At a given deflection δ_0, the elastic-plastic work necessary to deflect the specimen, dU, caused by a small increase in the crack length, da, is related to J as follows:

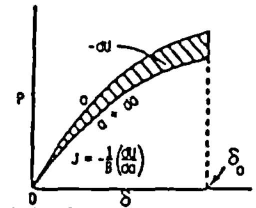
$$J = -\frac{1}{B}\frac{dU}{da} \qquad (1)$$

where B is the specimen thickness. Deflections, δ, must of course be measured colinear with the applied load vector. The procedure is similar to the compliance method of determining the linear elastic strain energy release rate, G.

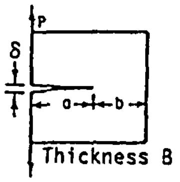
For deeply-notched compact tension and bend bar specimens, J is closely approximated [Ref. 74] by:

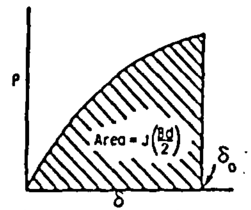
$$J = \frac{2}{Bb}\int_0^{\delta_0} P\,d\delta \qquad (2)$$

Crack closure effects during cyclic loading present the same sort of problems as they do in the application of linear-elastic fracture mechanics. The area above this point was used to calculate ΔJ as indicated in Fig. (3). Crack closure was measured using a clip gage attached to the side of the specimen to measure displacements across the crack tip, and from the load-versus-deflection hysteresis loops.



(a) Elastic-plastic Work Difference





(b) Rice et.al. approximation

Fig. 2 Determination of J from Load Versus Deflection Curves for Compact Tension

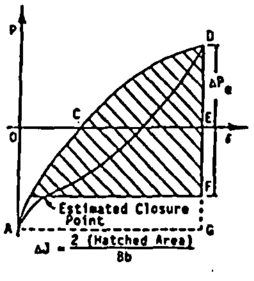


Fig. 3 ΔJ for Compact tensile Specimen Tests

Fig. (4) shows a plot of the resulting data which was characterized by large cyclic plastic deformations. Also shown is a least-squares-fit through these data which has the equation:

$$\frac{da}{dn} = 2.13 \times 10^{-8}\,(\Delta J)^{1.587}\text{, in/cycle} \qquad (3)$$

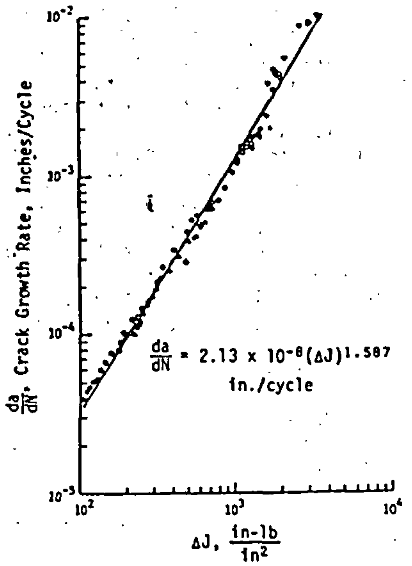


Fig. 4 ΔJ Fatigue Crack Growth Correlation For Gross Plastic Cyclic Deformations of Compact Tensile Specimens, Refs. [71, 72]

Ref. [76] reports fatigue crack growth rate data for large-scale plastic cycling of center cracked specimens of the same A533B steel used in the compact tension specimen tests. An analog control circuit was again used to achieve stable plastic deflections and compliance procedures were used to obtain ΔJ for the center cracked specimen. Fig. (5) shows the basic approach for the center cracked specimens, where J is determined by adding the elastic and plastic components.

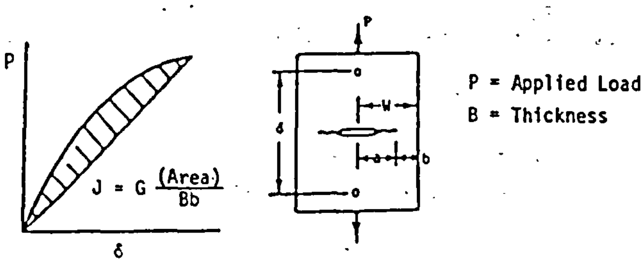


Fig. 5 Determination of J from Load Versus Deflection Curves for Center Cracked Specimlens

The elastic component of J is equal to G, the strain energy release rate, calculated from the applied load and crack length as if there were no plasticity. The elastic-plastic work difference was again applied only to that portion of the hysteresis loop during which the crack is opened. This is illustrated in Fig. (6).

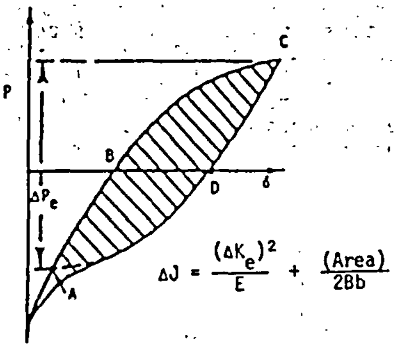


Fig. 6 ΔJ for the Center Cracked Specimen

The center cracked specimens were 0.50 in. thick with half width W = 0.967 in., and machined notch half-length a_0 = 0.50 in. Surface crack lengths were monitored visually with a low power traveling microscope at both ends, and in some cases at all four corners. Average crack length versus cycles data were thus obtained. Cyclic crack growth rates were determined, as in the case of the compact tensile specimens, using an incremental polynomial procedure, Ref. [78].

Fig. (7) shows a compilation of the resulting fatigue crack growth data obtained on the center cracked specimens under plastic loading conditions. The line shown in Fig. (7) is the same as the correlation shown in Fig. (4) for the compact tensile specimen data.

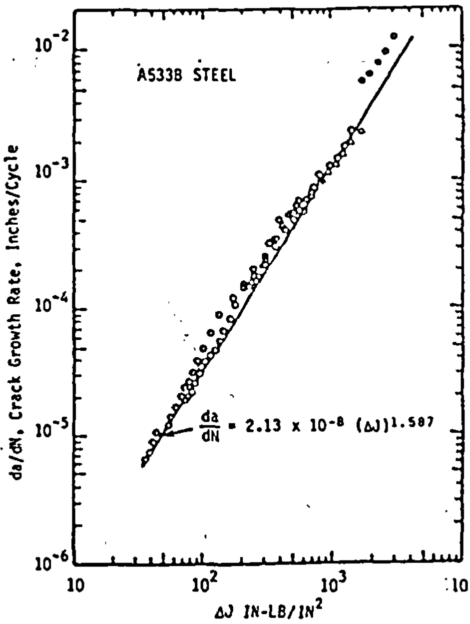


Fig. 7 Fatigue Crack Growth Rate Versus Cyclic ΔJ Data for Elastic-Plastic Tests on Center Cracked specimens, Ref. [76]

As a further extension of geometric parameters, data from Ref. [71] on oversize linear elastic compact specimens shown in Fig. (8) is useful. The same material was used in these tests as in the compact tension specimens and center cracked specimens. The dimensional parameters have the same proportions as the standard ASTM fracture toughness specimen parameters (ASTM Test for Plane Strain Fracture Toughness of Metallic Materials (E399-74). The dimensions of these large specimens are four times those of the specimens used in the compact tensile specimens of Fig. (4). Stress intensity ranges were determined from the applied load range and crack length by the usual methods of linear elastic fracture mechanics. The polynomial expression of ASTM Method E399-74, which is accurate out to a/W = 0.7, was used. These stress intensities were converted to J values for the plane strain conditions. The results are shown in Fig. (8) where the correlation line is the same as that in Figs. (4) and (7).

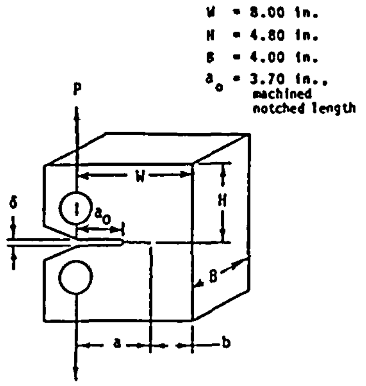


Fig. 8 Oversize Compact Specimen Parameters

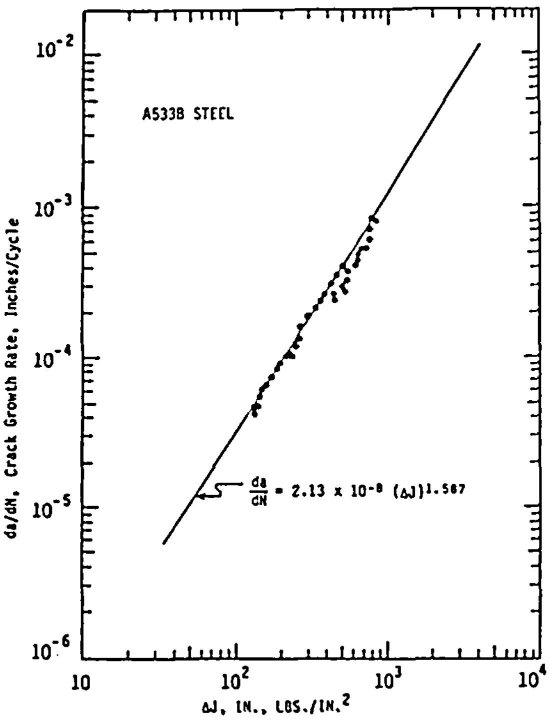


Fig. 9 Linear Elastic Test Results for Oversize Compact specimens, Ref. [71]

Figs. (4), (7) and (9) show that the J-integral concept, applied over a range of elastic-plastic fatigue crack growth rates, gives successful correlations including large-scale plasticity effects. We conclude that experimental data on A533B demonstrate that the J-integral is a valid geometry and size-independent correlation parameter for the elastic-plastic fatigue crack growth rates encountered in low-cycle fatigue. Considerable elastic-plastic crack growth testing has been carried out on other structural steels of interest (see for example, Refs. [79] through [84]). Jablonski, [Ref. 84], found an excellent correlation for the fatigue crack growth using the J-integral range for 304 Stainless Steel and for HY 100 material. Existing test results strongly support the use of ΔJ as a correlating parameter which is independent of size, geometry and loading conditions.

Existing da/dn vs. ΔK correlations based on tests in the elastic regime, can be converted to ΔJ using the elastic relation:

$$J_e = \frac{K^2}{E_1}, \text{ where } \begin{array}{l} E_1 = E \text{ for plane stress} \\ E_1 = E/(1 - \nu^2) \text{ for plane strain} \end{array} \quad (4)$$

In applying the ΔJ approach for elastic-plastic conditions:

$$J = J_e + J_z + J_p \quad (5)$$

where J_z is a plastic zone correction to J_e, and is usually small.

The J_p integral is developed in Refs. [85 through 93], and the EPRI Manual [Ref. 90] contains finite element results for many cases. In applications to cyclic crack growth, the mathematics are greatly simplified by expressing the plastic strains in the cyclic stress-strain curve by a power law, so that the total stress strain relation has the form:

$$\varepsilon = \sigma/E + (\sigma/A)^n \quad (6)$$

where n is the strain hardening coefficient.

For such a stress-strain relation, the plane stress and plane strain values of J_p have the form:

$$J_p = h(n, g)\sigma_1\varepsilon_p a \quad (7)$$

where σ_1 = nominal (linearized) stress

ε_p = nominal plastic strain

a = crack length

h(n, g) = geometry and strain hardening coefficient dependent function (differs for plane stress and plane strain)

The J-integral of Eq. (5) is then directly usable in the evaluation of known flaws, including combined thermal and mechanical cycling beyond yield. The da/dN vs. ΔK correlations from elastic cycling tests are the only crack growth data required, since elastic-plastic crack growth correlations based on the total J of Eq. (5)

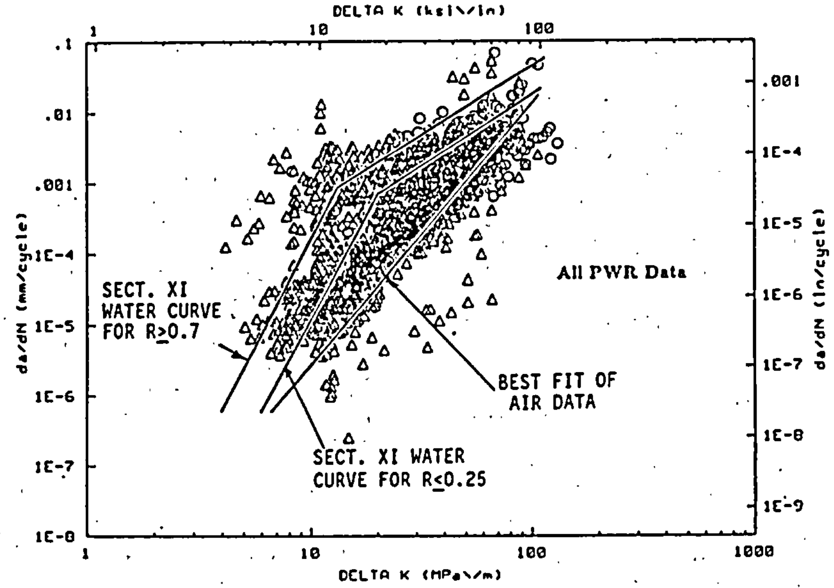


Fig. 10 Fatigue crack Growth Rate Data for A533 and A508 steel in PWR Environments (From EDEAC, Ref. [95], Courtesy of E. D. Eason)

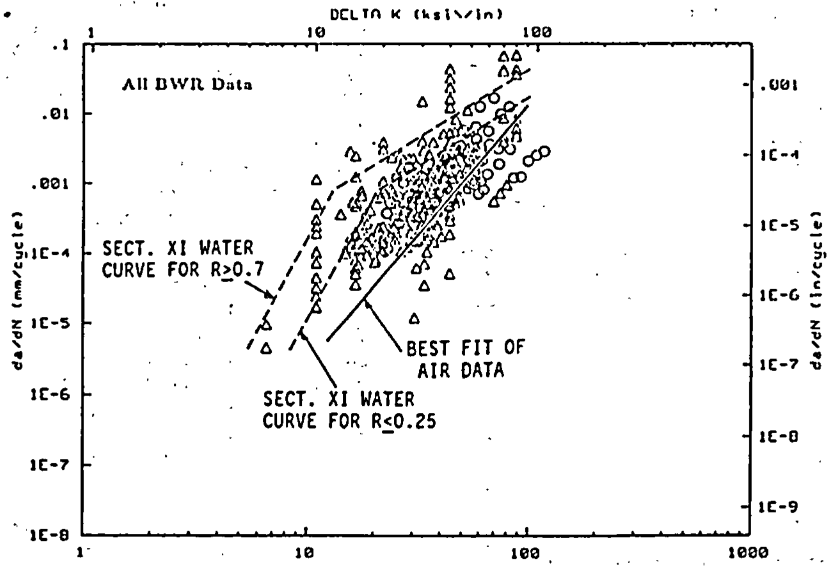


Fig. 11 Fatigue Crack Growth Rate Data for A533 and A508 Steel in BWR Environments (From EDEAC, Ref. [95], Courtesy of E. D.Eason)

coincides with the elastic correlations of Eq. (4). Thus, the use of ΔJ makes it possible to use elastic cycling test data to analyze elastic-plastic crack growth.

REACTOR WATER EFFECTS ON CRACK GROWTH OF FERRITIC STEELS

Fatigue crack growth rates (da/dn) of ferritic steels have been measured in a variety of environments including water chemistry conditions typical of both boiling and pressurized water reactor conditions. These measurements are based on linear elastic fracture mechanics and thus the fatigue crack growth rate is correlated to the range of the stress intensity factor Δk. Fig. (10) shows the available data in the pressurized water environment (PWR) and Fig. (11) shows the available data in the boiling water environment (BWR). This data is available through the "EPRI Database for Environmentally Assisted Cracking (EDEAC)," [Ref. 95].

These data show that both the PWR and BWR environment significantly increase the fatigue crack growth rates over the rates for air environments. The increase has been shown to be affected by frequency, mean stress as measured by the R-ratio defined as K_{MIN}/K_{MAX}, flow rate of the water past the crack, temperature of the water, and sulfur content of the steel. The current ASME Boiler and Pressure Vessel Code Section XI fatigue crack growth reference curves are based on a reasonable best-estimate of the worst combinations of these parameters.

These data are also providing the basis for more advanced mechanistic assessments which ultimately may allow for more realistic reference curve development. The Materials Properties Council Task Force on Crack Propagation Technology is monitoring an effort to develop such improved reference curves. Gilman [Refs. 15 and 96], VanDerSluys & Emanuelson [Ref. 97], Ford et al. [Ref. 13] and Scott et al. [Ref. 14], provide the theoretical and empirical underpinnings for this advanced work.

The most significant feature of this work is that although fatigue crack growth is a complicated function of material, environment, stress, geometry, and electro-potential mechanisms, a reasonably simple way to treat this process for design evaluations can be developed based on crack-tip strain rates (see Ref. [13]). Gilman [Ref.96] has developed models relating the da/dt crack growth rate in an aggressive environment such as water with the crack growth rate in a non- aggressive environment such as air. For practical applications in corrosion-assisted fatigue, it is convenient to think in terms of an expression of the form:

$$da/dN = C_1 C_2 \Delta K^n \qquad (8)$$

where the coefficient C_2 includes the time dependence in terms of cyclic frequency, and the R-ratio dependence. For the case of ferritic steels, C_2 is also highly influenced by the sulfur content and morphology of the steel.

Although these theories and data correlations are not yet fully developed, they do provide a basis for better quantifying the effects of water environment on the fatigue crack growth rates.

FATIGUE LIFE CORRECTIONS FOR CRACK GROWTH RATE DEPENDENCE

The S-N fatigue design curves of Sections III and VIII of the ASME Code [Ref. 99] are based on the number of cycles resulting in complete failure of the test specimens. Such test results include the crack initiation, propagation and fracture phases of failure. Because of the small size of the test specimens used (typically 0.3 in. diameter), crack initiation is often erroneously thought to play a predominant role in these tests. This is, in fact, only the case in the high-cycle regime. In the low-cycle regime, crack initiation occurs very early, at a small percentage of the fatigue life of the specimen. Thus, the crack propagation phase dominates the total fatigue life.

However, size effects are not as large as might be anticipated because most of the cyclic life of the specimen is spent growing short cracks (<30 mils). The crack growth rate near the end of the life of the specimen is so fast that the cycles-to-failure would not be greatly increased if the specimen diameter or component wall thickness were much larger.

O'Donnell [Ref. 1] has evaluated J-integrals for partial penny-shaped cracks growing in low-cycle fatigue test specimens. Thus, it is possible to determine the size of cracks that existed throughout the life of the specimen. This is done starting from the known cycles-to-failure in a given environment, and using the crack growth rate vs. ΔJ correlations for that environment to back-calculate the crack sizes. It is also possible to evaluate the effects that higher crack growth rates, that could occur in more hostile environments, would have on the total fatigue life.

Environmental effects on crack growth are basically time or strain-rate dependent. For the cycles of interest in operating components, such crack growth rate effects tend to peak in the intermediate ΔK or ΔJ range. For very short cracks at the low end of the ΔK or ΔJ range, corrosion effects tend to be small compared to strain cycling effects. This is illustrated by comparing the crack growth rate curves for reactor water environments with the air curves in Section XI of the ASME Code [Ref. 98]. Of course, where there are significant environmental effects on the threshold values or cycles-to-crack initiation, they can be added directly into the S-N fatigue life evaluation curves.

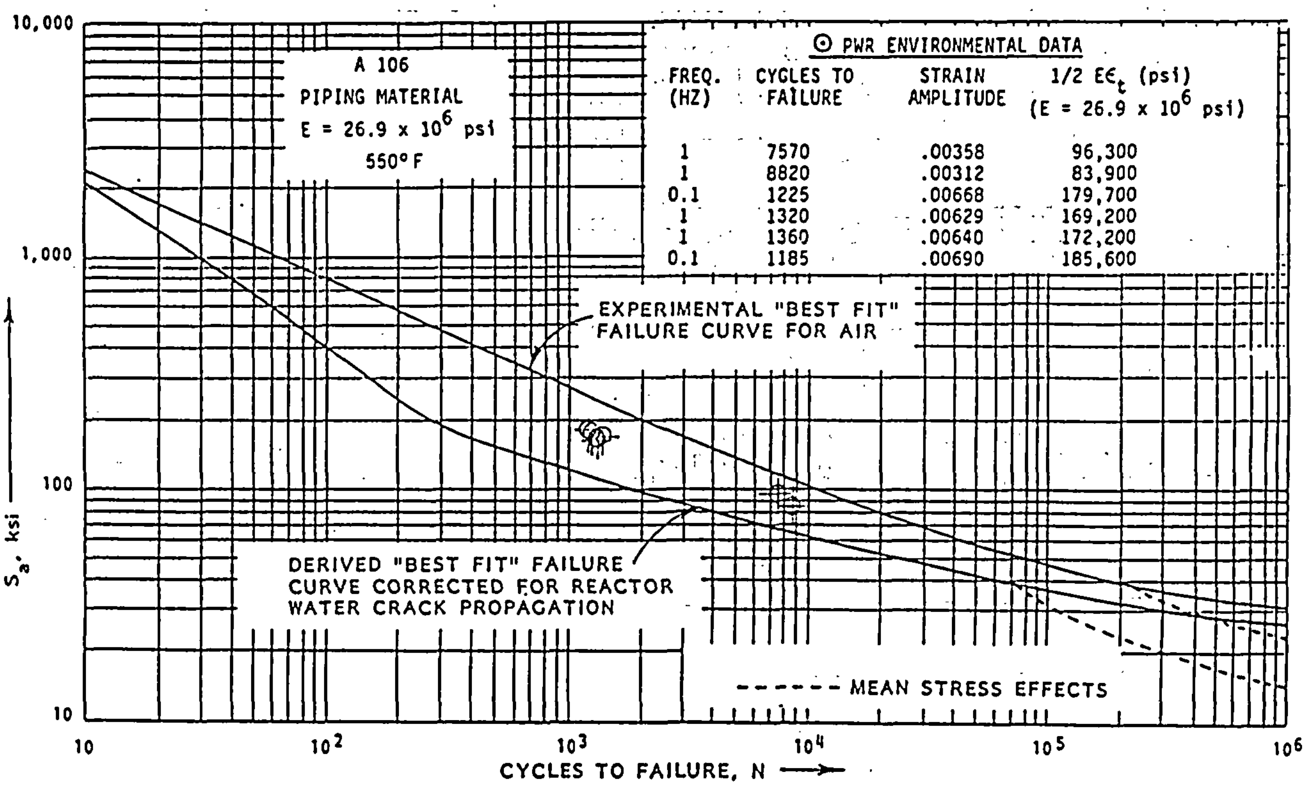
Taking SA 106-B piping material as an illustration of considerable importance, the upper curve in Fig. (12) shows the "best fit" failure curve used as a basis for the current Sections III and VIII design curves[3] [Ref. 99]. This failure curve was of course based entirely on data obtained in an air environment. Starting from the known cycles-to-failure at a selected strain amplitude, and integrating down the da/dN vs. ΔJ curve as described in Ref. [1], one can quantify the crack sizes that existed in the fatigue specimens at any point in their lives. The crack sizes obtained in such analyses on A533B were in good agreement with the crack sizes measured during interrupted fatigue tests by Dowling [Ref. 75]. The crack size integration is continued until it either reaches the beginning of life, or the point where the reactor water and air curves come together. Integrating forward from this point on the reactor water environmental curve to the failure conditions, one obtains a point on the lower curve shown in Fig. (12), which provides the cycles-to-failure accounting for the higher crack growth rates in reactor water per the Section XI crack growth curves.

A comparison of the upper and lower curves in Fig. (12) shows that reactor water environmental effects on crack propagation have a very significant effect on the cycles-to-failure in S-N fatigue. At intermediate strain ranges, there is a factor of about 5 difference in life.

After Fig. (12) was derived analytically, some directly relevant fatigue data was obtained under NRC sponsorship [Ref. 94]. Axial strain-controlled fatigue tests were run by MEA on smooth round bar specimens of ASME SA 106-B steel in PWR environments

---

[3] Of course, this curve was based on the data available when the design curves were being generated. Data from various steels were correlated using the Langer-Coffin Equation for various σ_{ult} values. While more accurate fatigue data is now available, this illustration was used to demonstrate the correspondence between the design criteria in Sections III and VIII, and the crack growth rates in Section XI of The Code.



⊙ PWR ENVIRONMENTAL DATA

FREQ. (HZ)	CYCLES TO FAILURE	STRAIN AMPLITUDE	1/2 $E\epsilon_t$ (psi) (E = 26.9 x 10^6 psi)
1	7570	.00358	96,300
1	8820	.00312	83,900
0.1	1225	.00668	179,700
1	1320	.00629	169,200
1	1360	.00640	172,200
0.1	1185	.00690	185,600

Fig. 12 A106 Piping Material Fatigue Failure Curve Corrected for Reactor Water Crack Propagation Compared with Fatigue Failure Curve in Air at 550°F

(1.0 ppb de-oxygenated water) at a stress ratio of -1.0. These tests were run at frequencies of 1.0 Hz and 0.1 Hz as indicated in the table on Fig. (12). The fact that these data points fall between the upper and lower curves is very encouraging,given the diversity of data and methods employed. Remember that the reactor water crack growth rate curves in Section.XI are intended to cover both BWR and PWR environments, including the slower cyclic rates encountered in component service. These results again indicate the repeatability, consistency and reliability of S-N technology in the evaluation of safe life.

The ASME Code applies a factor of 2 on stress amplitude or a factor of 20 on cycles-to-failure to the "best fit" failure curve (whichever is more limiting at each point) in order to produce a fatigue design curve. These factors were intended to cover scatter in the data, surface finish effects (the test specimens are polished), size effects and environmental effects. Since the latter are explicitly taken into account in the approach used herein, the factor of 20 on cycles was reduced[4] to a factor of 10 in order to retain a consistent safety margin. The resulting design curves for air and reactor water environments are shown in Fig. (13), along with the current Code fatigue design curve, [Ref. 99], applicable to this material.

CONCLUSIONS

The ΔJ approach can be used with existing elastic da/dN vs. ΔK environmentally assisted fatigue crack growth rate correlations for pressure vessel and piping steels to evaluate known flaws

---

[4] The factor of 2 on stress amplitude was not reduced because on-going activities in the ASME Subgroup on Fatigue Strength will extend the existing design curves beyond 10^6 cycles to cover vibration loading, and the curves will be lower at higher cycles.

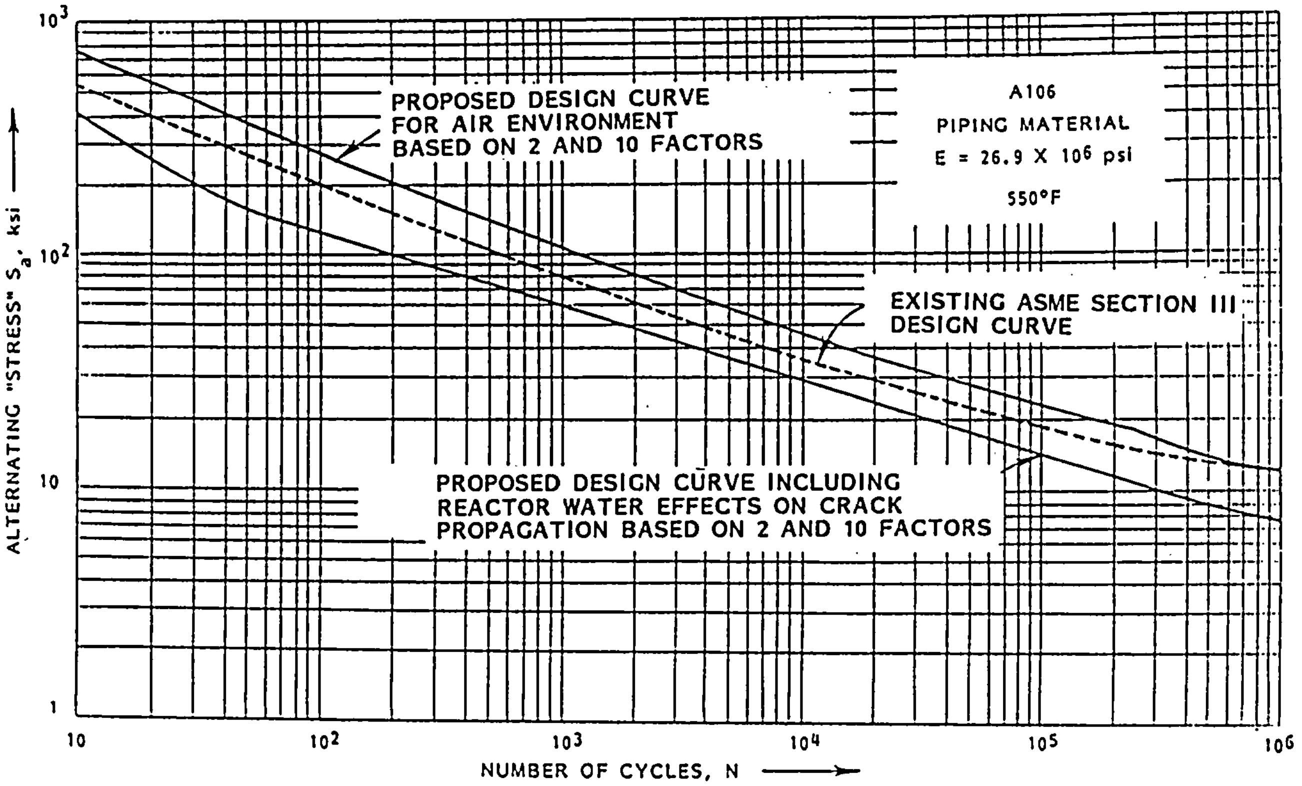


Fig. 13 Fatigue Evaluation Curves for A106 Piping Material in Air and Including Reactor Water Effects on Crack Propagation

subjected to combined thermal and mechanical cycling beyond yield.

The higher crack growth rates in reactor water environments vs. air in Section XI of the Code have a very significant effect on the S-N fatigue life as shown in Fig. (12). Fig. (13) shows a comparison of the corresponding fatigue design curves derived using safety margins consistent with the ASME Code with the existing Section III design curve for A106 piping steel. Cyclic rate effects, under consideration for inclusion in the crack growth rate curves of Section XI, can also be included in the proposed new design curves using the methods described herein. The inclusion of environmental and aging effects in S-N curves provides a sound basis for evaluating corrosion-assisted cumulative fatigue damage, and the residual operating life which would continue to meet ASME Code fatigue safety margins.

REFERENCES

1. O'Donnell, W.J., "Synthesis of S-N and da/dn Life Evaluation Technologies," ASME Paper 88-PVP-10, presented at the 1988 ASME Pressure Vessel and Piping Conf., Pittsburgh, Pa., June 19 - 23, 1988, to be published.
2. NUREG-1144, Nuclear Plant Aging Research (NPAR) Program Plan, prepared by the USNRC Division of Engineering Technology Office of Nuclear Regulatory Research, July 1985.
3. EPRI NP-4208, "The Longevity of Nuclear Power Systems," August 1985.
4. EPRI NP-2406, Project 1248-1, Final Report, "BWR Environmental Cracking Margins for Carbon Steel Piping," General Electric Company, May 1982.
5. EPRI NP-2775, Project 1325-1, Interim Report, "Corrosion Fatigue Characterization of Reactor Pressure Vessel Steels," The Babcock & Wilcox Company, December 1982.
6. NUREG/CR-1576, "A Review of Fatigue Crack Growth of Pressure Vessel and Piping Steel in High Temperature, Pressurized, Reactor-Grade Water," Cullen, W. H., Torronen, K., September 1980.
7. NUREG/CR-3294, "Fatigue Crack Growth Rates of A 508-2 Steel in Pressurized, High Temperature Water," Cullen, W. H., 1983.
8. NUREG/CR-2013, "Effects of Temperature on Fatigue Crack Growth of A 508-2 Steel in LWR Environments," Cullen, W. H., Torronen, K., Kemppainen, M., April 1983.
9. NUREG/CP-0044, "The Influence of Water Chemistry on Fatigue Crack Propagation in LWR Pressure Vessel Steels," Proceedings of IAE Specialists' Meeting on Subcritical Crack Growth, Cullen, W. H., USNRC Conf. Proceedings, May 1983.
10. NUREG/CR-4121, MEA-2053, "The Effects of Sulfur Chemistry and Flow Rate on Fatigue Crack Growth Rates in LWR Environments," Cullen, W.H./MEA, Kemppainen, M., Hanninen, H., Torronen, K./TRC, February 1985.

11. NUREG/CR-4422, MEA-2078, "A Review of the Models and Mechanisms for Environmentally Assisted Crack Growth of Pressure Vessel and Piping Steels in PWR Environments," Cullen, W., Gabetta, G., Hanninen, H., December 1985.

12. NUREG/CR-4723, MEA-2173, "Application of a Two-Mechanism Model for Environmentally-Assisted Crack Growth," Gabetta, G., Cullen, W.H., October 1986.

13. Ford, F.P., "Current Understanding of the Mechanisms of Stress Corrosion and Corrosion Fatigue," ASTM STP 821, American Society for Testing and Materials, 1984, pp. 32-51.

14. Scott, P.M., Tomkins, B., and Foreman, A.J.E., "Development of Engineering Codes of Practice for Corrosion Fatigue," Journal of Pressure Vessel Technology, 105, August 1983, p. 255.

15. Gilman, J.D., "Application of a Model for Predicting Corrosion Fatigue Crack Growth in Reactor Pressure Vessel Steels in LWR Environments," ASME PVP Vol. 99, Predictive Capabilities in Environmentally Assisted Cracking, November 1985.

16. NUREG/CR-4723, MEA-2173, "Application of a Two-Mechanism Model for Environmentally-Assisted Crack Growth," Gabetta, G., Cullen, W.H., October 1986.

17. Gangloff, R.P. and Wei, R.P. "Small Crack-Environment Interactions: The Hydrogen Embrittlement Perspective," Small Fatigue Cracks, edited by Ritchie, R.O. and Lankford, J., The Metallurgical Society, Inc., 1986.

18. Cottis, R.A., "The Corrosion Fatigue of Steels in Saline Environments: Short Cracks and Crack Initiation Aspects," Small Fatigue Cracks, edited by Ritchie, R.O. and Lankford, J., The Metallurgical Society, Inc., 1986.

19. Turnbull, A. and Newman, R.D., "The Influence of Crack Depth on Crack Electrochemistry and Fatigue Crack Growth," Small Fatigue Cracks, edited by Ritchie, R. O and Lankford, J., The Metallurgical Society, Inc., 1986.

20. Ford, F.P. and Hudak, S.J., Jr., "Potential Role of the Film Rupture Mechanism on Environmentally Assisted Short Crack Growth," Small Fatigue Cracks, edited by Ritchie, R.O and Lankford, J., The Metallurgical Society, Inc., 1986.

21. Cullen W.H. and Torronen, K., "A Review of Fatigue Crack Growth of Pressure Vessel and Piping Steels in High Temperature, Pressurized Reactor Grade Water," Naval Research Laboratory Report 4298, September 19, 1980.

22. International Atomic Energy Agency Symposium on Subcritical Crack Growth in Reactor Pressure Vessel Materials, Freiburg, Federal Republic of Germany, May 13-15, 1981.

23. Bamford, W.H., "Technical Basis for Revised Reference Crack Growth Rate Curves for Pressure Boundary Steels in LWR Environment," Journal of Pressure Vessel Technology, 102, November 1980, p. 433.

24. VanDerSluys, W.A. and Emanuelson, R.H. "Cyclic Crack Growth Behavior of Reactor Pressure Vessel Steels in Light Water Reactor Environments," ASME Crack Growth Symposium, New Orleans, December 1984.

25. VanDerSluys, W.A., and Emanuelson, R.H. "The Effect of Sulfur Content on the Crack Growth Rate of Pressure Vessel Steels in LWR Environments," Second International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Monterey, California, September 1985 (to be published).

26. Cullen, W.H., Kemppainen, M., and Torronen, K., "The Effects of Sulfur Chemistry and Flow Rate on Fatigue Crack Growth Rates in LWR Environments," USNRC Report NUREG/CR-4121, February 1985.

27. Jones, R.L., "Overview of International Studies on Corrosion Fatigue of Pressure Vessel Steels," Paper No. 170, National Association of Corrosion Engineers, Corrosion 84 Conference, New Orleans, April 1984.

28. EPRI NP-2775, "Corrosion Fatigue Characteristics of Reactor Pressure Vessel Steels," Dec. 1982.

29. Pearson, S., "Initiation of Fatigue Cracks in Commercial Aluminum Alloys and the Subsequent Propagation of Very Short Cracks," Engineering Fracture Mechanics, Vol. 7, No. 2, 1975, pp. 235-247.

30. Gilman, J.D., and Jones, R.L., "EPRI-Sponsored Research on the Influence of Reactor Environments on Fatigue Crack Growth," ASME Paper 82-PVP-22, June 1982.

31. Tice, D.R., "A Review of the UK Collaborative Program to Test the Effects of Mechanical and Environmental Variables on Environmentally Assisted Crack Growth of PWR Pressure Vessels," presented at European Federation of Corrosion, Conference on Environment-Sensitive Cracking, Munich, September 1984.

32. Cullen, W.H., Proceedings of IAEA Specialists' Meeting on Subcritical Crack Growth, NUREG/CP-0044, 1983.

33. Scott, D.M. and Truswell, Corrosion Fatigue Crack Growth in Reactor Pressure Vessel Steels in PWR Primary Water," Jrnl. of Pres. Vessel Tech., Vol. 105, August 1983.

34. Cullen, W.H., Torronen, K., Kemppainen, "Effects of Temperature on Fatigue Crack Growth of A508-2 Steel in LWR Environment," NUREG/CR-3230, 1983.

35. Bamford, W.H., Jacko, R.J., and Ceschini, L.J., "Environmentally Assisted Crack-Growth Technology," NUREG/CR-3744, 1984.

36. Cullen W.H., "Fatigue Crack Growth Rates of Low-Carbon and Stainless Piping Steels in PWR Environment," NUREG/CR-3945, 1985.

37. Cullen, W.H., "Proceedings of the Second IAEA Speacilists' Meeting on Subcritical Crack Growth," NUREG/CP-0067, 1986.

38. Ford, F.P., "Status of Research on Environmentally Assisted Cracking in LWR Pressure Vessel Steels," Proceedings ASME Pressure Vessel and Piping Conference, San Diego, CA, June 1987.

39. Nagata, N., Katada, Y., "Effects of Environmental Factors on Fatigue Crack Growth Behaviors of A533B Steel in BWR Water," Trans., 9th International Conference on SMiRT, Vol. F, LWR Pressure Components, 1987, p 167.

40. Tice, D.R., "Assessment of Environmentally Assisted Cracking in PWR Pressure Vessel Steels," Trans., 9th Annual International Conference on SMiRT, Vol. F, LWR Pressure Components, 1987, p 245.

41. Amzallag, C., Maillard, J.L., "Fatigue Crack Growth Behavior of Different Stainless Steels in Pressurized Water Reactor Environments," Trans., 9th International Conference on SMiRT, Vol. F, LWR Pressure Components, 1987, p 173.

42. Bamford, W.H., Wilson, I.L., "Quantitative Measures of Envoironmental Enhancement for Fatigue Crack Growth in Pressure Vessel Steels," Trans., 9th International Conference on SMiRT, Vol. F, LWR Pressure Components, 1987, p 137.

43. Ford, F.P., Andresen, P.L., "Modeling of Environmentally Assisted Cracking in the Stainless Steel and Low Alloy Steel/Water Systems at 288°C,"

Trans., 9th International Conference on SMIRT, Vol. F, LWR Pressure Components, 1987, p 259.

44. Kitagawa, H., Komai, K., Nakajimi, H., Higuchi, M., "Testing Round Robin on Cyclic Crack Growth of Low and Medium Sulfur A533-B Steels in LWR Environments," Trans., 9th International Conference on SMIRT, Vol. F, LWR Pressure Components, 1987, p 155.

45. Takeda, N., Hishida, M., Kikuchi, M., Hasegawa, K., Suzuki, K., "Crack Growth Study on Carbon Steel in Simulated BWR Environments," Trans., 9th International Conference on SMIRT, Vol. F, LWR Pressure Components, 1987, p. 161.

46. Rossheim, D. B., and Markl, A,R,C., "The Significance of, and Suggested Limits for, the Stress in Pipe Lines Due to the Combined Effects of Pressure and Expansion," Trans. ASME, Vol. 62, 1940, pp. 443-464.

47. Markl, A.R.C.,"Fatigue Tests of Welding Elbows and Comparable Double-Miter Bends," Trans. ASME, Vol. 69, 1947, pp. 869-879.

48. Markl, A.R.C. and George, H.H.,"Fatigue Tests on Flanged Assemblies," Trans. ASME, Vol. 72, 1950, pp. 77-87.

49. Markl, A.R.C.,"Fatigue Tests of Piping Components," Trans. ASME, Vol. 74, 1951, p. 126.

50. Dubuc, J., and Welter, G., "Investigation of Static and Fatigue Resistance of Model Pressure Vessels," The Welding Journal, 35 (7), Research Supplement 329-s to 337-s, 1956.

51. Welter, G., and Dubuc, J. "Fatigue Resistance of Simulated Nozzles in Model Pressure Vessels," ibid 36 (6) Research Suppl. 271-s to 274-s, 1957.

52. Kooistra, L.F., and Lemcoe, M.M., "Low Cycle Fatigue Research on Full-Size Pressure Vessels," Welding Research Supplement, July 1962, p. 297-s.

53. Pickett, A.G., and Grigory, S.C., "Prediction of the Low Cycle Fatigue Life of Pressure Vessel," Transactions of the ASME, Journal of Basic Engineering, December 1967, pp. 858-870.

54. NUREG/CR-0325, "Relevance of Fatigue Tests to Cold Leg Piping," Mayfield, M.E., Rodabaugh, E.C. and Eiber, R.J., Battelle Columbus Laboratories, 1978.

55. NUREG/CR-3543, ORNL/NSIC-216, "Survey of Operating Experience from LER's to Identify Aging Trends," Status Report.

56. NUREG/CR-4279, "Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety Related Piping and Components of Nuclear Power Plants, Phase I Study," February 1986.

57. NUREG/CR-3818, SAND 84-0374, Report of Results of Nuclear Power Plant Aging Workshops, May 1984.

58. NUREG/CR-3819, EGG-2317, "Survey of Aged Power Plant Facilities," June 1985.

59. Rice, J.R., J. Appl. Mech., 35, 1968, p. 379.

60. Rice, J.R., "Treatise on Fracture," Liebowitz, H., ed., 2, Academic Press, New York, 1968, p. 191.

61. Rice, J.R., "A Path Independent Integral and the Approximate Analysis of Strain Concentration by Notches and Cracks," Journal of Applied Mechanics, Transactions ASME, 35, June 1968.

62. Hutchinson, J.W., "Singular Behavior at the End of a Tensile Crack in a Hardening Material," Journal of the Mechanics and Physics of Solids, 16 (1), 1968, pp. 13-31.

63. Begley, J.A. and Landes, J.D., "The J-Integral as a Fracture Criterion," ASTM STP 514, American Society for Testing and Materials, Philadelphia 1972, pp. 1-20.

64. Landes, J.D. and Begley, J.A., "The Effect of Specimen Geometry on J_{Ic}," ASTM STP 514, American Society for Testing and Materials, Philadelphia, Pa., 1972, pp. 24-39.

65. Drucker, D.C. and Rice, J.R., "Eng. Fract. Mech.," 1, 1970, p. 577.

66. Rice, J.R. and Sorenson, E.P., J., "Mech. Phys. Solids," 26, 1978, p. 163.

67. Rice, J.R. and Rosengren, G.F., "Plain Strain Deformation Near a Tip in a Power-Law Hardening Material," Journal of the Mechanics and Physics of Solids, 16 (1), 1968, pp. 1-12.

68. Begley, J.A. and Landes, J.D., "The J-Integral as a Fracture Criterion," Fracture Toughness, Proceedings of the 1971 National Symposium on Fracture Mechanics, Part II, ASTM STP 514, American Society for Testing and Materials, Philadelphia, Pa., 1972, pp. 1-20 and pp. 24-39.

69. Broek, D., "Elementary Engineering Fracture Mechanics," 2nd Edition, Sijthoff & Noordhoff, Alphen aan den Rijn, the Netherlands, 1978.

70. Sehitoglu, H. and Morrow, J., "Characterization of Thermo-Mechanical Fatigue," Thermal and Environmental Effects in Fatigue: Research-Design Interface, PVP-71, American Society of Mechanical Engineers, New York, 1983, pp. 93-110.

71. Paris, B.C., Bucci, R.J., Wessel, E.T., Clark, W.G.,Jr., and Mager, T.R., "In Stress Analysis and Growth of Cracks," Proceedings of the 1971 National Symposium on Fracture Mechanics, Part I, ASTM STP 513, American Society for Testing and Materials, 1972, pp. 121 and 141-176.

72. Dowling, N.E. and Begley, J.A., "Fatigue Crack Growth During Gross Plasticity and The J-Integral," Mechanics of Crack Growth, ASTM STP 590, American Society for Testing and Materials, Philadelphia, Pa., 1976, pp. 82-103.

73. Rice, J.R., "Fracture - An Advanced Treatise," Vol. 2, Mathematical Fundamental, Academic Press, N.Y., 1968, pp. 191-311.

74. Rice, J.R., Paris, P.C., Merkle, J.G., "Progress in Growth and Fracture Testing," ASTM STP 536, 1973, pp. 231-245.

75. Dowling, N.E., "Crack Growth During Low-Cycle Fatigue of Smooth Axial Specimens," Cyclic Stress-Strain and Plastic Deformation Aspects of Fatigue Crack Growth, ASTM STP 637, American Society for Testing and Materials, Philadelphia, 1977, pp. 97-121.

76. Dowling, N.E., "Geometry Effects and the J-Integral Approach of Elastic-Plastic Fatigue Crack Growth," Cracks and Fracture, ASTM STP 601, American Society for Testing and Materials, Philadelphia, Pa., 1976, pp. 19-32.

77. Dowling, N.E., "Fatigue-Crack Growth Rate Testing at High Stress Intensities," Flaw Growth and Fracture, ASTM STP 631, American Society for Testing and Materials, Philadelphia, Pa., 1977, pp.139-158.

78. Clark, W.G., Jr., and Hudak, S.J., Jr., Journal of Testing and Evaluation, Vol. 3, No. 6, Nov. 1975, pp. 454-476.

79. Mowbray, D.F., "Derivation of a Low-Cycle Fatigue Relationship Employing the J-Integral Approach to Crack Growth," Cracks and Fracture, ASTM STP 601, American Society for Testing and Materials, Philadelphia, Pa., 1976, pp. 33-46.

80. El Haddad, M H., Dowling, N.E., Topper, T.H. and Smith, K.N., "J-Integral Applications for Short Fatigue Cracks at Notches," International Journal of Fracture, 16 (1), 1980, pp. 15-30.

81. Jaske, C.E. and Begley, J.A., "An Approach to Assessing Creep/Fatigue Crack Growth," Ductility and Toughness Considerations in Elevated Temperature

Service, MPC-8, American Society of Mechanical Engineers, New York, 1978, pp. 391-409.

82. Sadananda, K. and Shahinian, P., "Application of J-Integral Parameter to High Temperature Fatigue Crack Growth in Cold Worked Type 316 Stainless Steel," International Journal of Fracture, 15, 1979, R81-84.

83. Taira, S., Ohtani, R., and Komatsu, T., "Application of J-Integral to High-Temperature Crack Propagation Part II - Fatigue Crack Propagation," Journal of Engineering Materials and Technology, 101, 1979, pp. 162-167.

84. Jablonski, D.A., "An Experimental Study of the Validity of a Delta-J Criterion for Fatigue Crack Growth," Instron Corporation Report, Third ASTM International Symposium on Nonlinear Fracture Mechanics, Knoxville, Tenn., October 1986.

85. Lamba, H.S., "The J-Integral as Applied to Cyclic Loading," Engineering Fracture Mechanics, Vol. 7, 1975, pp. 693-703.

86. Parks, D.M., "Elastic-Plastic Continuum Mechanics of Fatigue Crack Growth," Final Report in Grant AFOSR-77-3424, Dept. of Engineering and Applied Science, Yale University, New Haven, Conn., Jan. 1978.

87. Tanaka, L., "The Cyclic J-Integral as a Criterion for Fatigue Crack Growth," Int'l. Jrnl. of Fracture, Vol. 22, 1983, pp. 91-104.

88. Wuthrich, C., "The Extension of the J-Integral Concept to Fatigue Cracks," Int'l. Jrnl. of Fracture, Vol. 20, No. 2, 1982, pp. R35-37.

89. Goldman, N.L. and Hutchinson, J.W., "Fully Plastic Crack, The Center-Cracked Strip Under Plane Strain," International Journal of Solids and Structures, 11, 1975, pp. 575-591.

90. Kumar, V., German, M.D. and Shih, C.F., "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI NP-1931, Project 1237-1, Electric Power Research Institute, Palo Alto, Calif., July 1981.

91. He, M.Y. and Hutchinson, J.W., "The Penny-Shaped Crack and the Plane Strain Crack in an Infinite Body of Power-Law Material," Jnl. of Applied Mechanics, ASME, Vol. 48, No. 4, December 1981, pp. 830-840.

92. He, M.Y. and Hutchinson, J.W., "Bounds for Fully Plastic Crack Problems for Infinite Bodies," Second Int'l. Symp. on Elastic-Plastic Fracture, Philadelphia, October 1981. To be published in an ASTM STP.

93. Trantina, G.G., de Lorenzi, H.G. and Wilkening, W.W., "Three-Dimensional Elastic-Plastic Finite Element Analysis of Small Surface Cracks," Engineering Fracture Mechanics, Vol. 18, No. 5, pp. 925-938, 1983.

94. Terrell, J.B., "Fatigue Strength of ASME SA 106-B Piping Steel in 288°C Air and PWR Environments," MEA Report to ASME Subgroup on Fatigue Strength, December 1987.

95. Mindlin, A.H., et al., "EPRI Database for Environmentally Assisted Cracking (EDEAC)," EPRI Report NP-4485, April 1986.

96. Gilman, J.D., "Further Development of a Model for Predicting Corrosion Fatigue Crack Growth in Reactor Pressure Vessel Steels," Jrnl. of Pressure Vessel Technology, Trans., ASME, Vol. 109, August 1987, pp. 340-346.

97. VanDerSluys, W.A. and Emanuelson, R.H., "Enhancement of Fatigue Crack Growth Rates in Pressure Boundary Materials due to Light-Water-Reactor Environments," SMiRT, 1987.

98. ASME Boiler and Pressure Vessel Code, Section XI, 1986 Edition, with 1987 Addenda, Article A-4000.

99. ASME Boiler and Pressure Vessel Code: 1986 Edition, with 1987 Addenda, Section III, Appendix I and Section VIII, Div.2, Appendix 5.

FATIGUE TRANSIENT COUNTS FOR PWRs:
A STUDY OF FIVE WESTINGHOUSE PLANTS

H. W. Massie, Jr.
C. B. Bond

The term "Plant Life Extension (PLEX)" as used in this paper originates from the definition of "Life Extension" contained in the Atomic Industrial Forum/National Environmental Studies Project-040 (AIF/NESP-040) and shall not be deemed to be a warranty or representation that such plant and/or equipment can be operated economically and safely for the initial license term or beyond. Rather the term "PLEX," which is based upon engineering judgments, technological developments, operating plant data, and analysis of such developments and data, attempts to predict the capability for continued operation of the plant and for the equipment beyond the licensing basis of such plant and/or equipment.

Abstract

Westinghouse has completed five life extension studies for various utilities. Each of these studies has included a historical review of operating transients and fatigue assessment of critical nuclear steam supply system components. Based on the analyses performed to date, the rate of accumulation of transient cycles for many postulated design transients is lower than that predicted 10 or 20 years ago in the design basis. However, other factors (such as unanalyzed transient conditions and possible changes in nuclear plant operating modes and code criteria over an extended lifetime) have potential to offset this reduction. The fact that many utilities are undertaking programs to deal with fatigue and other degradation modes well in advance of application for license extension provides reason for optimism in the nuclear industry that utility objectives for nuclear power plant life extension will be accomplished in a safe and effective manner. This paper presents a summary of the results of actual transient counts, fatigue accumulation, and major findings from a fatigue aging standpoint.

Introduction

A nuclear power plant is postulated to undergo thousands of cycles of each of several different transient events during its lifetime. Table 1 lists the types of transients typically specified for a Westinghouse pressurized water reactor (PWR). Transient, stress, and fatigue analyses are performed for critical components when the plant is designed, to assure that the components will maintain structural integrity when subjected to these postulated transient events.[1-5] The ASME Boiler and Pressure Vessel Code provides the criteria for such analyses. Also, utilities generally maintain records of operational occurrences of selected events to track conformance of plant operation with design. With increased interest in plant life extension among utilities has come concern about the adequacy of the fatigue design basis for a longer plant life. Plant life extension studies generally include a review of fatigue on critical plant components due to plant operating transients. The results and conclusions from five such transient and fatigue studies are summarized in the following sections.

Discussion of Study Results

As part of its Plant Life Extension (PLEX) program, Westinghouse has counted and evaluated transients on five different nuclear plants. For the first time, a comparison can be made among five different plants for the same vendor. Table 2 summarizes the ranges of parameters for the five plants of interest. The plants range in net power rating from 320 MWe to 781 MWe, two-, three-, and four-loop Westinghouse configurations, and from 13 to 18 years of plant operation, at the time of the study. The design life objective for fatigue evaluation purposes is either 30 or 40 years depending on the vintage of the plant. All plants studied were operated in a base load manner, which results in a lower number of fatigue-inducing cyclic plant load changes compared to the original design basis. This lower level of fatigue accumulation is one measure related to plant safety. The lifetime plant availability is of interest as a general indicator of the frequency of certain transient occurrences. Nuclear plants with higher cumulative availability would be expected to have experienced a lower number of unplanned lifetime transients, such as reactor trips.

Table 1
TYPICAL DESIGN TRANSIENT CYCLES
FOR WESTINGHOUSE REACTOR COOLANT SYSTEM
AND MAJOR NSSS COMPONENTS

Normal condition transients
- Plant heatup and cooldown (100°F per hour)
- Plant loading and unloading (5% of full power per minute)
- Small (10%) step load increase or decrease
- Large step load decrease with steam dump actuation (50% or 95%)
- Steady-state fluctuations

Abnormal condition transients
- Loss of load (without immediate turbine or reactor trip)
- Loss of power
- Partial loss of reactor coolant flow (that is, loss of one reactor coolant pump)
- Reactor trip from 100% power
- Feedwater cycling

Test/shutdown condition transients
- Turbine roll test
- Primary side hydro test (at 3105 psig)
- Secondary side leakage test
- Primary side leakage test (at 2485 psig)
- Secondary side hydrostatic test
- Refueling

Transients for RCS auxiliary systems
- Pressurizer spray actuation
- Pressurizer safety valve actuation
- Pressurizer relief valve actuation
- RTD bypass maintenance operations
- Charging flow shutoff and return to service
- Letdown flow shutoff and return to service
- Charging flow step increase or decrease
- Letdown flow step increase or decrease
- Charging and letdown flow shutoff and return to service
- High head safety injection actuation

The source of the transient data collection is also important. Generally, one of the following approaches was used in supplying transient data for the PLEX studies:

- Utility performed all data collection.
- Independent consultant or vendor collected data.
- Data were collected jointly by utility and consultant.

The variety of approaches resulted in broad differences among the historical transient data bases, some of which were more useful than others in establishing a complete picture of the transient and fatigue operating history of the plant. This fact indicates a need for criteria and guidance in establishment of a comprehensive and uniform method for transient data collection for PLEX and other purposes. The vendor owners' groups and technical societies are currently working toward establishing such approaches.

Table 2
COMPARISON OF KEY PARAMETERS IN FIVE WESTINGHOUSE PLANTS

Parameter	Plant A	Plant B	Plant C	Plant D	Plant E
	Value for Indicated Plant				
Net power rating (MWe)	781	582	320	485	520
Number of loops	3	4	2	2	2
Years of operation (as of study)	13	18	15	16	13
Design life objective (years)	40	30	30	40	40
Operational history	Base-load	Base-load	Base-load	Base-load	Base-load
Lifetime availability factor	65.1	82.0	64.1[a]	79.8	81.5

---

a. Includes 4-year shutdown

Based on the data collected, Table 3 presents a comparison of design and actual transients for five Westinghouse nuclear plants based on recently completed plant life extension (PLEX) studies. The first column lists eight typical design transients, including their severity. In most cases, the actual transients experienced were not as severe as the design transients. Moreover, in most cases, only a transient count was obtained for the study. Relying solely on transient counts results in added conservatism in any accompanying fatigue evaluation based on operating transient data. The conservatism arises because many plant events are partial occurrences of the full design basis events counted. The reduction in unneeded conservatism for evaluation of fatigue for design basis transients can become particularly important for those locations having fatigue-inducing transients that were not considered in the design basis, such as surge nozzle and stratification transients.[6]

The second column lists the number of each transient allowed or typically specified in the design specification over a 40-year design life objective. The five Westinghouse life extension projects found the specified number of transients to be conservative in many cases, mainly because load follow operation was assumed in design. The plants in the study all operated in base load mode most of the time. There is therefore some basis for optimism for utility owners who will seek to obtain license renewals in the 1990s and beyond.

This optimism must be tempered, however. Utilities must note the assumptions on which it is based and evaluate their own specific cases carefully. The first assumption is that of continued base load operation. At present, many utilities base-load their nuclear units and load follow with fossil units. However, many fossil units are even older than the nuclear units. As these units are retired, nuclear plants may experience additional load follow cycling. Further, flexible operating options such as economic generation control are being implemented or considered by utilities. These operating modes can increase the fatigue experienced by plant components.

Second, although events related to load follow (such as heatups and load changes) are occurring at rates lower than the design basis, abnormal events (such as reactor trips) are sometimes occurring at higher rates. One plant described in Table 3, along with two others from more recent PLEX studies, are currently accumulating reactor trip counts at a rate such that the 40-year allowables will be used up before the end of design life. When the reduced severity of the trips is taken into account, the problem with the transient counts is less severe than it might appear. However, instances like this show the need for keeping good records of transient operation. Also, since a reactor trip can be one of the more severe events a plant experiences, the need for excellence in plant operation to achieve PLEX is highlighted.

Finally, the studies reported in this paper considered only design basis events. Several locations in nuclear plants are experiencing fatigue accumulation due to transients unanticipated at the time most plants were designed. In many cases, the life of a component can be shortened considerably from 40 years and, in a few cases, leaks have been reported (most notably in cases of stratification). These unanalyzed conditions certainly do not preclude life extension, but effective means, such as operation modifications, monitoring, inservice inspection, and repair, must be incorporated in a realistic life extension program for fatigue.

The remaining columns in Table 3 list the actual operating transients for each of the five nuclear plants. Also listed is the years of plant operation at the time of each study. Plant A experienced 63 heatups and cooldowns in 13 operating years, but these heatups and cooldowns were typically at 20°F to 30°F per hour; hence margin exists in the se-

Table 3
COMPARISON OF DESIGN AND ACTUAL TRANSIENTS FOR FIVE WESTINGHOUSE PWRs
(Selected Transients Only)

Typical Transient	Number of Transients in Design Specification[a]	Actual Transient Occurrences in Indicated Plant				
		Plant A (13 yr)	Plant B (18 yr)	Plant C (15 yr)	Plant D (16 yr)	Plant E (13 Yr)
Plant heatup at 110°F/hour	200	63[b]	27	48	42	36
Plant cooldownat 100°F/hour	200	63[b]	27	47	41	37
Plant loading at 5% of full power per minute	14,500 (varies)	190	186	Not reported	1,341	Not reported
Plant unloading at 5% of full power per minute	14,500 (varies)	190	186	Not reported	1,206	Not reported
Large step load reduction (typical 100% to 50%)	200	1	14	0	20	2
Loss of load (without immediate trip)	80	1	0	3	0	Not reported
Reactor trip from full power	400	170[c]	134[c]	14	42[c]	38
Primary side hydrostatic test (~3105 psig)	5	1	1	1	1	1

---

a. Typical
b. Heatups/cooldowns significantly <100°F/hour
c. Includes lower power trips

verity of the transients. The other plants experienced lower numbers of transients.

All plants except one (plant D) experienced a very low number of plant loadings and unloadings. The low numbers may reflect the difficulty of obtaining enough detailed information from the plant operating records. Particularly in the case of load changes, a detailed search must be conducted to obtain all the partial cycles, since load changes are more routine than other operations, such as heatup. Thus, the number of events is larger and the individual events stand out less sharply. In each of these studies, however, hundreds of man-hours were spent to obtain the data. Under the category of reactor trips, the totals for plants A, B, and D also include low-power trips (one-third to one-half of total) which contribute less to fatigue than full transients.

Table 3 also lists one primary side hydro test at 3105 psig, which is 1.25 times design pressure (P_D). This transient is counted as the hydro test performed prior to core fuel load (and of course prior to plant startup). In addition, some of the major reactor coolant system (RCS) primary components such as the reactor vessel have had a hydro test at 1.25 P_D performed in the shop. Thus the reactor vessel would actually have seen the equivalent of two of the higher-pressure hydro tests.

The total primary loop typically experiences one primary side hydro test (at 3105 psig). Every 10 years as part of Section XI requirements, the primary loop is tested at 1.1 times the plant primary operating pressure (1.1 × 2235 psig). In addition, this less severe hydrotest is performed after the pressure boundary has been broken – for example, after completion of refueling.

Figure 1 presents the total number of reactor trips for each of the five plants, plotted as a function of plant age. Also plotted are plant heatups. No general trend is noted for the trips. For the heatups, the older plants appear to have had fewer heatups per unit time. This may be a reflection of the plant availability and reflects less plant outage time. However, the trend is noted for only a set of five plants.

Data Evaluation

In a nuclear plant life extension study, transient data such as that in Table 3 are then evaluated by systems engineers for consistency and to specify transients used by the component groups evaluating fatigue. The rate of transient occurrences represents a rough indicator of an aging trend for fatigue. Table 4 presents, for plant E, the results of an evaluation for the specification of transients for PLEX. Again, the first column lists the transients of interest. The second column presents the specified number of transients for each of the transients. It should be noted that the specified number of plant loadings and unloadings is different from that in Table 2, because the information in this table is plant-specific rather than typical as in the preceding tabulation. The third column lists the actual number of transients experienced on plant E during 13 years of operation, and the next column is a straight-line projection out to 40 years of life for each of the transients. This projection assumes that the plant operating conditions for the next 27 years (out to 40 years) will be identical to those of the first 13 years of operating life. The last column is the recommended new design transient specification for a 60-year design life for plant E.

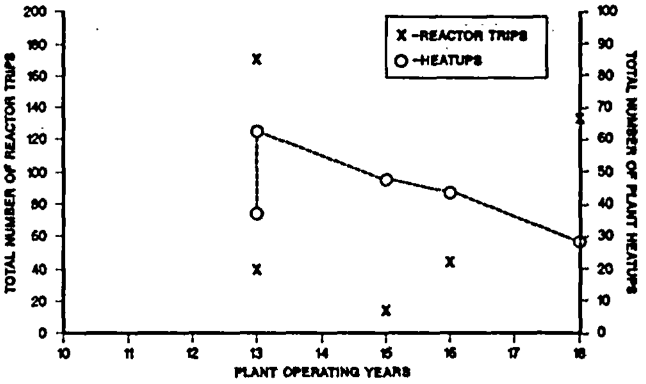


Figure 1. Number of Reactor Trips and Plant Heatups for Various Plants Versus Years of Operation

Specification of the number of transients for a 60-year design life objective is based on the actual transient history plus additional margins necessary to cover changes in future plant operations as well as to assure adequate plant safety margins. For plant E, the specified number of reactor trips for 60 years is now only 350 compared to an original design basis of 400 over 40 years (or 10 per year assumed). Because of the good operating record of plant E (high availability, low number of reactor trips), the specified number of trips for a life extension objective can now be reduced. This also illustrates the tie between life extension, availability (reactor trips), and plant safety margin.

Fatigue Usage Implications

Although transient event categories are important and widely understood, it should be noted that the actual fatigue accumulation due to actual operating transients remains the best measure of remaining life. In all the studies performed to date, the components for which substantial margins were not found were the pressurizer surge line

Table 4

COMPARISON OF PROPOSED 60-YEAR PLANT TRANSIENTS VERSUS 40-YEAR DESIGN AND ACTUAL TRANSIENTS (PLANT E)

Typical Transient	Design Number of Transients (40 yr)	Actual Number of Transients (13 yr)	Projected Number of Transients for 40 Years Based on Actual	Updated Design Number of Transients for 60-Year Life
Plant heatup at 110°F/hour	200	36	110	250
Plant cooldownat 100°F/hour	200	37	110	250
Plant loading at 5% of full power per minute	18,300	Not reported	N/A	6.250
Plant unloading at 5% of full power per minute	18,300	Not reported	N/A	6.250
Large step load reduction	200	2	6	200
Loss of load (without immediate trip)	80	Not reported	N/A	80
Reactor trip from full power	400	38	120	350
Primary side hydrostatic test (~3105 psig)	5	1	1	5

nozzles, the feedwater nozzles, the pressurizer spray nozzles and upper shell, and the auxiliary nozzles. Recent events, however, indicate other areas that are accumulating fatigue at rates greater than planned for in the design bases. These include selected areas in safety injection and residual heat removal piping. Data for these events are currently being evaluated.

It should also be noted here that the concept of fatigue usage (ASME Code, Section III) must be replaced by fatigue crack growth (ASME Code, Section XI) when flaws are present, as is the case in most real engineering components. Over the past 20 years, much has been learned about engineering materials in general and fatigue properties in particular. This new knowledge includes the effects of environments and thermal and irradiation aging. The ASME Code committees are currently developing means to incorporate this knowledge into the fatigue code criteria, such as fatigue usage and crack growth curves. Thus the trend will be to replace implicit factors for uncertainty with explicit treatment in revised Code criteria. It will be necessary to incorporate the effect of these potential changes in long-term assessments of fatigue for life extension.

Finally, fatigue is an irreversible mechanism. Unlike embrittlement, for example, for which annealing techniques are available, there is at present no known metallurgical technique for countering the effects of fatigue.

For these reasons, carefully conceived transient monitoring and inservice inspection programs are essential to any prudent program for plant life extension. No detailed fatigue reanalyses were performed in the present PLEX studies because of the cost involved at this stage. Later, as the plants age, it is anticipated that updated fatigue analyses would be performed for the most critical locations and for submittal of a license renewal application in the 1990s.

Major Findings

The results of this study provide some notes of optimism, along with some realistic perspectives for those utility owners attempting to manage transient and fatigue accumulation for life extension. The major findings are as follows:

- The actual operating transients are in many cases less severe than those specified in the design basis, for example,
  - Heatup and cooldown rates are generally less than 100°F per hour.
  - Power reactor trips are typically a mix of low- and high-power trips.
- Previously unanalyzed conditions will play an increasing role in the evaluation of fatigue life.
- Current assumptions for plant operation (those that formed the basis for this paper) may change in the future. These potential changes must be considered in the design of PLEX programs.
- Except for reactor trips, the transient counting rate is less than the design basis rate assumption (40 years) by a factor of two for the base load units in these studies.
- Plants with high availability have fewer unplanned transients (for example, fewer reactor trips).
- A need exists to identify the number of reactor trips for various power levels. One rule of thumb is that trips below approximately 20 percent power have little or no fatigue impact for many of the primary reactor components.
- For plant D, the detailed Westinghouse study identified more transient loadings and unloadings than were reported for other plants. Other plants may need to assess this area carefully.
- As expected, plants have not experienced many large step load reductions, loss of load transients, step load increases/decreases, and the like. However, margin must be maintained in case the frequency of these transients increases with plant age.
- Plants may have to modify the transient design basis to include other transients:
  - Feedwater cycling
  - Inadvertent RCS depressurization
  - Inadvertent auxiliary spray
  - Core overpower excursions

  This modification, which Westinghouse expects will be limited to a few critical areas of the primary system components, can be incorporated in any fatigue reanalyses required for license renewal.
- Much knowledge about fatigue has accumulated in 20 years. This accumulation is likely to result in code criteria changes in the years ahead.
- Obtaining data on transient severity probably requires computerization. A substantial effort (hundreds of man-hours per plant) was required to obtain the data in this paper. Some data definition is lost without a computerized system.
- Fatigue margin can be utilized as one indicator of plant safety margin.

Options to Monitor Fatigue Aging and Mitigate its Effects

The utility owner has several options available to monitor aging and/or mitigate effects associated with fatigue. First, it will be prudent to maintain detailed records to track the plant transients. If transient severity (pressure and temperature) data are later required, a computerized approach is most effective. As plants age, increased (but selective) inspection of areas with the highest calculated fatigue is likely to be necessary. On the other hand, perhaps other inspection requirements can be relaxed. Keeping informed on other plant experiences is essential in incorporating the effects of unanalyzed conditions. The last option is to make repairs and or refurbishments in selected areas of high fatigue. A well-designed program of historical evaluation, monitoring, inservice inspection, and fatigue evaluation will help to ensure maximum component life for fatigue considerations.

References

1. Barson, J.M., and Rolfe, S.T., *Fracture and Fatigue Control in Structures*, Prentice-Hall, 1977.
2. Forrest, P.G., *Fatigue of Metals*, Pergamon Press, 1962.
3. McEvily, A.J., Jr., and Teteloman, A.S., *Fracture of Structural Materials*, John Wiley & Sons, 1967.
4. Dieter, G.E., *Mechanical Metallurgy*, McGraw-Hill, 1976.
5. Osgood, C.C., *Fatigue Design*, John Wiley & Sons, 1970.
6. Bond, C.B., and Palusamy, S.S., "Perspectives on Nuclear Power Plant Transient and Fatigue Monitoring," *Proceedings*, 1988 ASME PVP Conference, PVP-Vol. 138.

ELASTOMER SHELF LIFE: AGED JUNK OR JEWELS?

Bruce M. Boyum and Jerral E. Rhoads
Washington Public Power Supply System

ABSTRACT

The shelf life of elastomeric products used in the Nuclear Industry is typically based on military standards (MIL-HDBK-695C [1] or MIL- STD-1523A [2]). Recently, data became available on naturally aged O-rings that were over 30 years old. An evaluation of this data is presented to demonstrate the conservatism of current guidelines.

KEYWORDS

Elastomer, Shelf Life, Rubber, Polymer, Seals, O-Rings, Aging, Storage Conditions, Physical Properties

I. BACKGROUND

Elastomeric materials are natural or synthetic rubber products that are used in a variety of equipment related applications such as seals, diaphragms, hoses, belts, and valve seats. An elastomer is a material that is capable of recovering from large deformations quickly and forcibly [17]. With time, the physical properties of these materials change. This is particularly evident under severe service conditions. Eventually the material can change to the point that the elastomer no longer performs its intended design function and replacement is necessary. Consequently, a spare parts inventory is maintained for periodic replacement of these materials. Elastomer properties change even under storage conditions and, consequently, a shelf life is typically assigned to them. The shelf life defines how long the material can be kept in storage before it should be disposed of.

Utilities typically obtain shelf life limitations from the component supplier or establish it themselves through the use of military standards MIL-STD-1523A or MIL-HDBK-695C. Component suppliers also typically rely on these same standards. A partial listing of the shelf life limitations specified by MIL-HDBK-695C is given in Table 1.

Table 1

MIL-HDBK 695C Shelf Life Limitations

Type Rubber	Common Or Trade Name	Shelf Life
Fluorocarbon	Fluorel, Viton	20 yrs
Ethylene/Propylene	Ethylene/Propylene Copolymer	5 to 10 yrs
Butadiene/ Acrylonitrile	Nitrile, Buna N	3 to 5 yrs
Silicone	Silicone	20 yrs
Butadiene/Styrene	SBR	3 to 5 yrs
Polychloroprene	Neoprene	5 to 10 yrs
Isobutylene/ Isoprene	Butyl	5 to 10 yrs
Polyester Urethane	Urethane	3 to 5 yrs

MIL-HDBK-695C addresses most common elastomers and uses except specialized applications such as adhesive, tapes, rubber-asbestos packing, rubber-cork gaskets, and unvulcanized rubber. This standard is meant to control elastomer age following receipt by the government and is considered a guide only. There are no specific storage environmental restrictions provided.

MIL-STD-1523A specifies a shelf life of eight (8) years for elastomeric hose and ten (10) years for O-rings. It addresses only elastomers that are resistant to petroleum based fluids, primarily nitrile. A maximum storage temperature of 125°F and protection from sunlight, circulating air, ozone and foreign materials are specified. This standard is meant to control elastomer age prior to receipt by the government. This standard is based on an aerospace requirement that elastomers should have a shelf life of at least eight (8) to ten (10) years to facilitate procurement and product development durations. The shelf life is not based on known limitations of the elastomer and consequently, it does not provide a technical basis for limiting shelf life.

The exact basis for MIL-HDBK-695C has not been determined by the authors. Our review of the literature [4], [5], [6] from the time period during which the standard was being developed (prior to 1964) indicates that natural aging data was available on elastomers for an aging duration of five (5) to seven (7) years. Figures 1 through 5 indicate what typical elastomer property changes looked like over this time interval. These elastomers were judged to be acceptable after five (5) to seven (7) years of natural aging but with the trend in property change, it was reasonable to believe shelf life limitations were necessary. This paper will review what happens after this first seven (7) year period and relate it to shelf life.

II. ELASTOMER COMPOUNDING AND CURING

The compounding of an elastomer can have an effect on the degree of physical property change during storage and service conditions. These factors include [9], [10]:

a) base polymer used
b) filler used
c) percent composition of ingredients
d) curing system used
e) additives such as antioxidants, softeners, plasticizers, etc.

The curing or vulcanization process transforms the material from weak and plastic like into a strong and elastic product. In the process, the rubber loses its tackiness and develops better resistance to degradation by heat, light and aging processes. The curing takes the polymer chains and crosslinks them, thus developing the physical properties.

There are a variety of curing systems that are used [11]. A common curing process involves sulfur in which the sulfur atoms form a bond between carbon atoms on adjacent polymer chains thus crosslinking the molecules. Other curing systems promote formation of radicals to form the carbon to carbon bonds on adjacent polymer chains. Peroxide cure is typical of this process. A nonsulfur cure is required on saturated rubbers since there are no double carbon bonds available to accommodate sulfur bonding. Crosslinks that involve carbon to carbon bonds are considered quite stable. Also sulfur bonds with only one or two sulfur atoms are considered more stable than longer chain sulfur bonds.

The curing agents that are added to the polymer may remain in the compound following product formation. This can lead to continuation of curing following the vulcanization process. When this occurs, additional property changes may be observed as the elastomer continues to cure.

III. ELASTOMER PHYSICAL PROPERTIES

The most commonly measured physical properties of elastomers in aging tests are hardness, tensile strength, ultimate elongation, modulus, and compression set [12]. These properties are not as useful in determining an elastomer's acceptability from an engineering sense as similar properties are for steel. Also the normal inter-relationship between properties when applied to steel are not necessarily true with elastomers. For example, steel hardness and tensile strength are directly related but for elastomers, this may not be true. These physical properties are typically used only as a guide for the acceptability of a material for an application or in comparing the relative usefulness of two materials for a given application. The following is an indication of how these properties are used when applied to seal materials and are fairly representative of most elastomer applications [3], [13].

1. Hardness (resistance to surface indentation)

   a) low durometer (soft material)

      o flow (seal) easier - desireable for rough surfaces

   b) high durometer (harder material)

      o greater resistance to flow (extrusion) - desireable at high pressure
      o helps prevent extrusion, nibbling, and spiral failure in dynamic applications

2. Tensile Strength (force per unit of original cross-sectional area required to rupture specimen)

   a) not ordinarily an important factor in O-ring design if a compound has over 1000 psi tensile
   b) not indicator of extrusion resistance
   c) sometimes used to indicate relative change when materials contacted with fluids

3. Ultimate Elongation (stretch expressed as percent of original length at the time of specimen rupture)

   a) defines stretch tolerated during installation
   b) sometimes used to indicate relative change in materials contacted with fluids

4. Modulus (tensile stress at a given elongation)

   a) indicates extrusion resistance
   b) indicates likelihood of material to recover from a peak overload or localized force

5. Compression Set (percent deflection an elastomer fails to recover after a set squeeze for a given time and temperature)

   a) indicates materials sealing capability
   b) large changes can occur in static seal application without significant effect

When creating a test program to establish a materials acceptability after shelf aging, the test parameters and acceptance criteria must be selected. Unfortunately there is no one parameter that readily lends itself to this application [8]. The parameter chosen must:

a) adequately represent the aging of the material and,
b) relate to the function of the material

Since no one parameter typically meets these criteria, a variety of material properties must be evaluated together to assess the acceptability of aged materials.

IV. AGING PROCESS

When an elastomer ages, there are three (3) mechanisms primarily involved [7]:

a) scission - The process of breaking molecule bonds. Typically due to ozone attack, UV light, or radiation.
b) crosslinking - The process of creating molecular bonds. Typically due to oxygen attack, heat, or curing.
c) compound ingredient evaporation, migration, mutation, etc.

Of these aging processes, scission and crosslinking have the major impact on physical property change. The following property changes are related to scission:

a) increased elongation
b) decreased tensile strength
c) decreased modulus

Crosslinking results in the reverse of the above changes.

A material in storage can be susceptible to a variety of different reactions such as ozone or oxygen attack and continued curing. The effect on the material property change as a result of these reactions can be different depending on whether the reaction results in scission or crosslinking. Also the rate at which each reaction occurs is temperature dependent. This paper is based on natural aging data and consequently, the reactions occurring in the tested material should be representative of those in material in storage under similar conditions.

V. NATURAL AGING TEST DATA

The Rubber Manufacturer's Association (RMA) initiated an elastomer natural aging study in 1957 [5]. This study involved seven (7) elastomer manufacturers that measured the physical properties (hardness, modulus, elongation, and tensile strength) of a variety of elastomers over a ten (10) year period.

The study included the following polymers:

a) neoprene
b) nitrile
c) viton
d) butyl
e) SBR
f) urethane
g) ethylene propylene
h) silicone
i) acrylate

The study concluded that after ten (10) years, the elastomers were acceptable. At this point, the RMA discontinued the study, however, two of the initial participants kept the remainder of the original samples in storage. In 1987, further testing was done on these samples that were then 24 and 31 years old. Table 2 provides a representative sample of the initial and final properties for each type of elastomer in the study. The properties reported in Table 2 are the median value of either three (3) or five (5) measurements. The measurements were made in accordance with ASTM Standard, D1414 [14].

The O-rings had been stored both loose in boxes or packaged in heat sealed, polyethylene lined, paper bags with only O-rings of a like material in each container. The RMA concluded that there was no significant improvement in shelf aging characteristics for packaged/sealed O-rings over those stored in boxes [5]. The storage temperature range was approximately 65°F to 85°F with an average estimated to be 75°F.

TABLE 2

ELASTOMER PROPERTY CHANGE

Polymer	Age	Hardness [1] (PTS)	Tensile St (PSI)	Modulus [2] (PSI)	Ultimate Elongation	Compression [3] Set
Nitrile	0 yrs	70	1800	900	200	75
	31 yrs	90	2290	1800	150	80.9
Neoprene	0 yrs	70	2150	450	340	57
	31 yrs	82	1940	1100	175	73.8
Butyl	0 yrs	71	1490	450	350	87
	31 yrs	70	1630	550	300	97
Viton	0 yrs	75	2010	450	320	37.2
	26 yrs	79	2040	600	310	45
SBR	0 yrs	68	1500	550	200	41
	31 yrs	77	1400	850	150	47
Urethane	0 yrs	73	3200	600	475	93
	30 yrs	80	3000	1160	265	103
Silicone	0 yrs	70	800	-	85	10
	31 yrs	72	825	-	60	11.7
Acrylate	0 yrs	70	1300	470	210	37
	31 yrs	64	1360	300	350	45.5
Ethylene/	0 yrs	81	1600	685	216	-
Propylene	24 yrs	79	1680	741	221	-

[1] Durometer - Shore A
[2] Modulus at 100% elongation
[3] Compression set after 70 hrs. at 250°F

VI. MATERIAL PROPERTY CHANGES

Graphical presentation of material property changes over the life of the study for a nitrile, neoprene, and SBR elastomer is provided in Figures 1, 2, 3, 4, and 5. These three (3) materials were chosen since they typically represented the most severe property changes of the elastomers tested. As indicated by Table 2, most of the other elastomers exhibited little change by comparison with these three.

There is no exact criterion against which to evaluate the acceptability of a material whose physical properties have changed. However, to assist in judging the significance of a change, material tolerances which are utilized by a large U.S. manufacturer of O-rings [3] are provided in Table 3. This manufacturer generally considers a material within specification if it is within these tolerances for the property specified. This is not an acceptance criterion for this paper but is included as a guide for judging significance of the changes.

Table 3

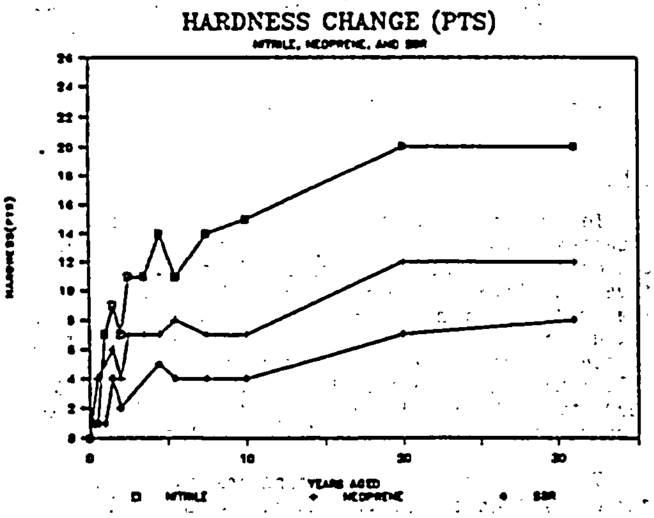
Material Tolerances

Property	Tolerances
Hardness	±5 PTS
Tensile Strength	±15%
Modulus	±25%
Elongation	±20%

The property changes observed over the duration of the test period (typically 31 years) are presented and discussed below.

Hardness

Figure 1

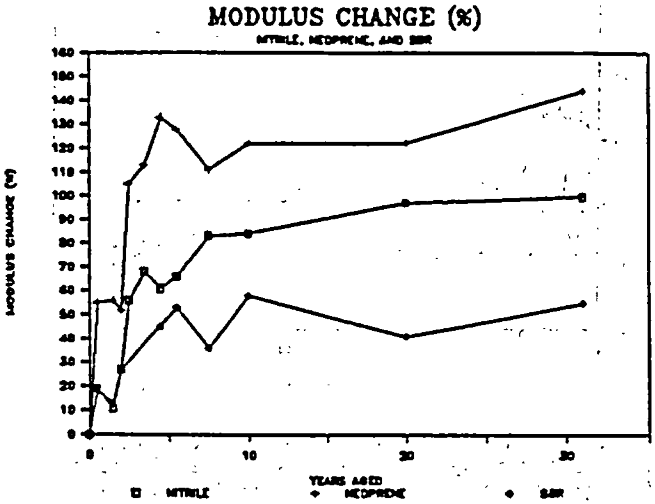


In general, the materials got harder as they aged. The nitrile and neoprene in Figure 1 had a rapid increase in hardness for the first five (5) to eight (8) years and then increased gradually after that. This behavior was typical for materials tested that exhibited any appreciable hardness change.

A few materials got softer (see Table 2). The worst case was acrylate which softened to approximately the lower tolerance for a new material (Table 3).

Modulus

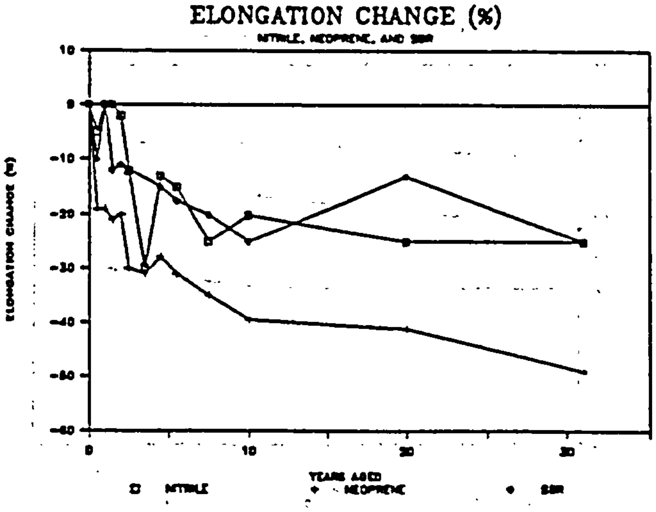
Figure 2



The modulus for all three (3) compounds increased rapidly over the first five (5) years and then increased gradually after that. As indicated in Table 2, an increase in modulus with age was seen in all materials tested except acrylate. The acrylate modulus decreased approximately 36% versus a tolerance of 25% (reference Table 2 and 3).

Elongation

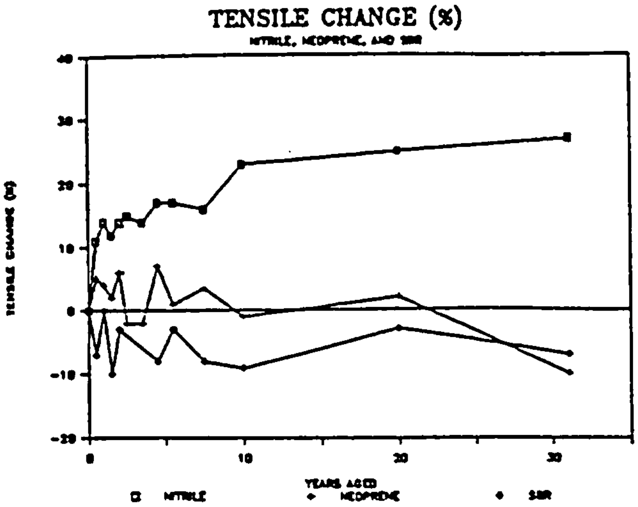
Figure 3



Elongation typically decreased as the materials aged. As indicated in Figure 3, the elongated change in nitrile, neoprene, and SBR was rapid in the first three (3) to ten (10) years followed by a gradual decrease. At the end of the test, the nitrile was capable of 150% elongation, the neoprene 175%, and the SBR 150% (see Table 2). A decrease in elongation was observed in all materials except ethylene propylene and acrylate (see Table 2). The ethylene propylene elongation remained virtually unchanged (+2%) while that of the acrylate increased 67%.

Tensile Strength

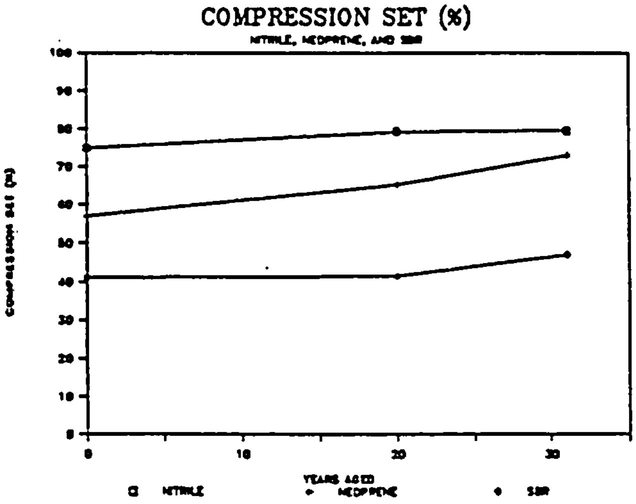
Figure 4



The tensile strength of the nitrile in Figure 4 increased rapidly for the first three (3) to ten (10) years then stabilized for the duration of the test. Most of the nitriles tested behaved in this manner. The remaining materials tested (Figure 4 and Table 2) showed relatively minor tensile change, ending within the tolerance given in Table 3 of ±15%. As indicated by Table 2, the direction of tensile change varied, some increasing slightly and others decreasing. It is noted also that all materials that had decreasing tensile strength, ended with a strength in excess of 1000 psi.

Compression Set

Figure 5



The compression set testing was conducted after the material was compressed for 70 hours at 250°F. The compression set testing was only done at 0, 20, and 31 years. Consequently, it is not possible to establish how rapidly this property changed during the initial twenty years. As indicated in Figure 5, the compression set for these SBR, neoprene, and nitrile compounds changed very little over 31 years. All materials tested exhibited increased compression set as they age as indicated by Table 2. The only materials that exhibited high compression set had changes of only 12% (see Table 2).

VII. DISCUSSION OF TEST RESULTS

The test data indicates that the material property typically changes in the following general trends (when the change is of any significance):

- hardness increases
- tensile strength increases
- elongation decreases
- compression set increases

As indicated in Section IV, the above property changes are associated with crosslinking of the polymer. Crosslinking in turn is typically due to heat, oxygen, or curing. Materials with the most severe property changes, such as nitriles, typically exhibit the above changes in two stages. Initially, there is a relatively rapid property change which peaks out in the first five (5) to ten (10) years. This is then followed by a much more gradual property change. The temperature and oxygen levels have remained effectively constant with regard to the test samples. Crosslinking due to oxygen is occurring over the entire test period. However, the rapid change in properties in the first five (5) to ten (10) years is due to curing controlled crosslinking. This is followed by gradual property changes associated with oxygen promoted crosslinking only. This conclusion is consistent with the findings of the Rubber Manufacturer's Association [5].

The curing in the initial five (5) to ten (10) years of the elastomer's life can result from:

a) excess curing agent remaining following the initial high temperature curing process and,
b) when sulfur curing is utilized, sulfur bridges of multiple sulfur atoms in series can be formed. With time, these bridges can free sulfur atoms to create additional crosslinks between carbon chains. The elastomers in this test with the more severe property changes, were all sulfur cured.

The following is an assessment of the changes observed in each property.

Hardness

In general, a harder material is a more acceptable material within the range of hardness observed in the test specimens. This is especially true for high pressure and dynamic applications. The most common high pressure failure mechanism is extrusion or nibbling while spiral failure is a common dynamic failure. A harder material helps alleviate these problems [3]. A harder material can lead to leakage but generally this would be an inconvenience and not an equipment failure.

The major concern with hardness is related to materials that soften with time. This can lead to less resistance to pressure (extrusion) and unsatisfactory performance in dynamic application. The few materials that softened during the test, changed very little with the worst case being approximately equal to the new material tolerance (Table 3).

Modulus

Modulus is a good indication of extrusion resistance with a high modulus indicating greater resistance. Consequently, most compounds improved in this regard [5]. One acrylate compound decreased from a modulus of 470 psi to 300 psi. While this decrease of 36% is somewhat beyond the new material tolerance of 25%, the other properties of tensile strength, compression set, and hardness are all acceptable. Note also that this material was developed for low pressure automotive transmission and power steering applications where extrusion resistance (i.e., modulus) is not a concern.

Elongation

Good design practice recommends no more than 5% elongation on assembled O-rings [3]. Elongation is a property that is of interest primarily when considering the amount of stretch needed to install the elastomer. Consequently, the percent change in elongation is not nearly as important as the actual elongation capability remaining. The amount of stretch needed for installation varies from application to application and is difficult to generalize about. ASTM D2000 [15], however, does recognize minimum elongation capabilities down to 50% elongation for both silicone and nitriles. These two polymers had the lowest elongation of the materials tested and were greater than the 50% minimum.

Tensile Strength

Other than the nitriles, the tested materials typically did not have a tensile change greater than the new material tolerance (±15%, Table 3). All the nitrile compounds tested increased in tensile strength, the maximum being approximately +30%. As indicated in Section III, tensile strength is not a design consideration unless it is under a minimum value of 1000 psi. Consequently, the nitriles changed in a beneficial manner with regard to tensile strength. This is further indicated by ASTM Standard D2000 [15] which provides material properties for rubber products. It provides minimum tensile strength requirements but no upper limits.

Compression Set

Most materials had compression set values that were well below the 100% allowable [16]. The two that approached or exceeded 100% set (urethane and butyl) were being tested at a temperature (250°F) in excess of their service temperature of 212°F for urethane and 225°F for butyl [13]. This resulted in material softening or melting and corresponding material shrinkage which contributed to the compression set exhibited. Consequently, these tests are not representative of the compression set anticipated for appropriate service conditions for these two elastomers. Note also that the compression set changed less than 12 percent for these two materials, from new to aged condition. This indicates that aging did not appreciably impact their compression set.

VIII. CONCLUSIONS

Natural aging tests indicate that where there is a significant property change in a polymer, it occurs within the first five (5) to ten (10) years after initial formulation/curing. It appears this results primarily from a continuation of the curing process. The material changes that occur following this period are very gradual and are attributed to material aging.

Current shelf life practices are usually based on military standard MIL-HDBK-695C and MIL-STD-1523A. These standards prescribed a shelf life of three (3) to 20 years for the elastomers addressed here. Consequently, the elastomers are being stored through the period where most physical property change is occurring. The elastomers are then being successfully used in service which indicates the property changes that occur in storage are acceptable. Since the elastomers change very gradually following the military storage period, an extended storage period seems justified.

An important aspect of extended life is the storage conditions used. The test samples were stored such that they were not exposed to ultraviolet light or radiation. They were not near a source of ozone, and they had a controlled environment where the average temperature was not excessive. Further, they were stored in an unstressed condition and only like elastomers were stored together.

It is concluded that the shelf life for elastomers can be substantially longer than provided by MIL-HDBK-695C and MIL-STD-1523A provided that:

1. the elastomers are stored in a controlled environment and manner similar to that presented in this paper and,
2. the elastomers are purchased from a reputable manufacturer who maintains a quality controlled manufacturing process to ensure repeatability of elastomer formulation.

Areas of Further Study

Further work is being sponsored by the Electric Power Research Institute (EPRI) related to the data presented in this paper. An additional data point is being obtained for the elastomers after they have aged one more year (32 years total for most of the materials). Included also will be an assessment of data scatter and an attempt to simulate material aging data by accelerated aging. The EPRI report will contain all the test data not presented in this paper.

This paper is based on testing of O-rings. It is anticipated that the aging of other elastomeric products will proceed in a similar nature to what was discussed. However, due to the nature of the applications, there are several areas that require further review for age susceptible elastomers:

1. Hose, belts, and diaphragms - these applications introduce the failure mechanisms of flex fatigue and creep which are not generally a concern in seal application. The aging of elastomers and its impact on flex failure or creep is not addressed here.
2. Valve seats - Elastomeric valve seats are typically used on small valves where leakage is a concern. It is recognized that elastomers generally harden as they age thus making sealing more difficult. The severity of hardening for more age susceptible elastomers such as nitriles and its impact on valve sealing capabilities is not included here.

3. Seals stored installed - Elastomers that are stored in a stressed condition (stretched, compressed, etc.) may age at a more rapid rate.

Also some elastomers continue to cure for five (5) to ten (10) years. When an elastomer cures (crosslinks) while in a stressed configuration, it will tend to stay in that configuration, i.e., if it is compressed, it will exaggerate its compression set. This paper does not assess the severity of this phenomenon.

VIII. REFERENCES

1. MIL-HDBK-695C, Military Standardization Handbook Rubber Products: Recommended Shelf Life, Department of Defense, March 27, 1985.
2. MIL-STD-1523A, Age Controls of Age-Sensitive Elastomeric Material, Department of Defense, February 1, 1984.
3. Parker O-Ring Handbook ORD 5700, Parker Seal Group, 1982.
4. Department of the Air Force Technical Manual AFML-TR-67-235, Literature Survey on the Effects of Long-Term Shelf Aging of Elastomer Materials, March 1968.
5. Rubber Manufacturer's Association Long-Term Shelf Aging Tests Data Report, March 1966.
6. Parker Seals Technical Report TR P397B, Evaluation of Aged 10-13N, April 1, 1964.
7. Presray Manual, Aging Characteristics of Presray Seal and Gasket Material, March 1986.
8. EPRI NP-1558, A Review of Equipment Aging Theory and Technology, Electric Power Research Institute, September 1980.
9. R. Barbarin, "Selecting Elastomeric Seals for Nuclear Service," Power Engineering, December 1977, pp. 58-61.
10. SYN-76-1290, Chlorobutyl Rubber Compounding and Applications, Exxon Chemicals, 1976.
11. M. Morton, Ed; Rubber Technology, New York, Van Nostrand Reinhold, 2nd. Edition, 1973.
12. The Language of Rubber, E.I. duPont de Nemours and Co., Revised 1980.
13. O-Ring Handbook, Precision Rubber Products Corporation, March 1986.
14. ASTM Designation D1414, Standard Methods of Testing Rubber O-Rings, Reapproved 1982.
15. ASTM Designation D2000, Standard Classification System for Rubber Products in Automotive Applications, 1980.
16. AI-AEC-13146, Permeation, Leakage and Compression Set Testing of Elastomeric Seals for LMFBR Use, Atomics International, April 2, 1975.
17. ASTM Designation D1566-85b, Standard Terminology Relating to Rubber, June 1985.

IX. ACKNOWLEDGEMENT

The authors wish to acknowledge Mr. Howard Gillette of Precision Rubber Products and Mr. Don Bowman of Parker Seals for providing much of the data and technical guidance upon which much of this work is based.

X. BIBLIOGRAPHY

Bruce M. Boyum

Received the BS degree in Chemical Engineering from Michigan Technological University in 1969 and the MS degree in Nuclear Engineering from the University of Arizona in 1971.

From 1971 to 1978, he worked for Bechtel Power Corporation as a Mechanical Engineer on nuclear power plant design. In 1978, he joined Washington Public Power Supply System in the Engineering Group overseeing the design of WNP-2, a nuclear power station. He subsequently joined Burns & Roe, then Cygna Energy Services in support of engineering work on WNP-2. He has worked for Washington Public Power Supply System in the Equipment Engineering Department since 1986.

Mr. Boyum is a registered professional Mechanical Engineer in the State of California and is a member of Tau Beta Pi.

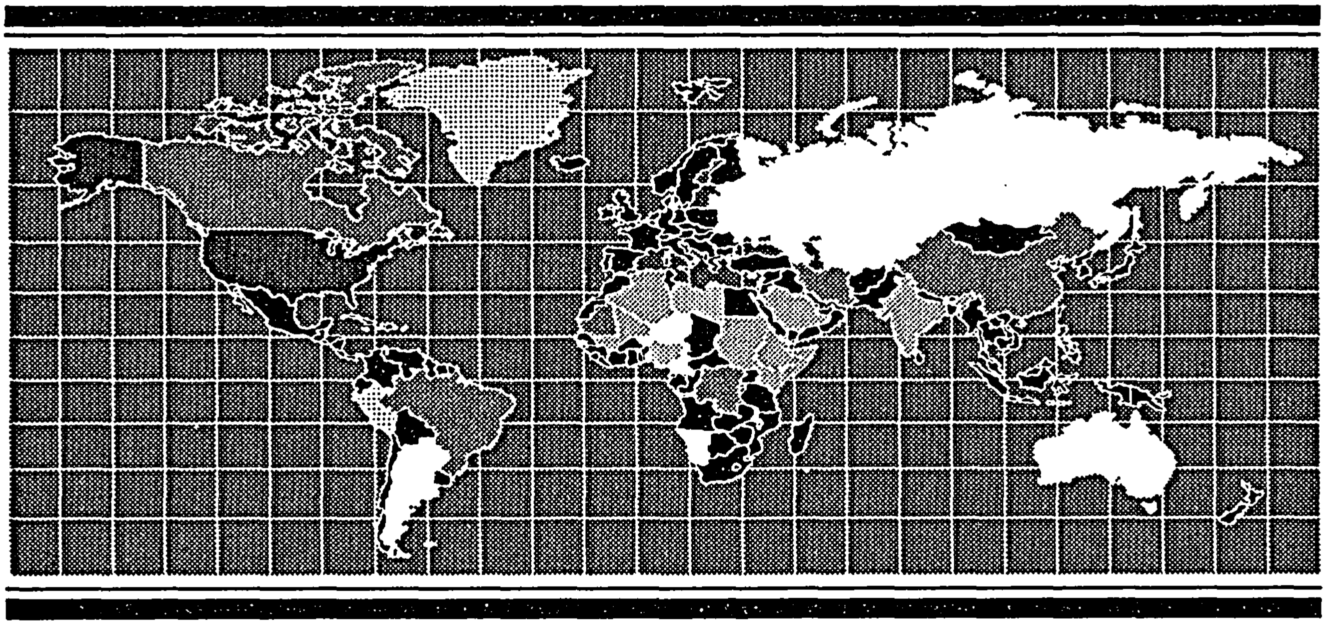
Jerral E. Rhoads

Mr. Rhoads is a registered professional Electrical Engineer in the State of Washington. He received the BS degree in Electrical Engineering from California State University at Sacramento in 1973.

From 1973 to 1975, he worked for Bechtel Power Corporation as an Electrical Engineer on nuclear power plant design and construction management.

In 1976, he joined Washington Public Power Supply System as Project Electrical Engineer. During his 12 years with the Supply System, he has assumed corporate management of the equipment qualification program. He also serves EPRI as the utility liason to the NRC research programs.

His IEEE Society activities include Chairman of WG 2.2, IEEE STD 334 "Standard for Qualifying Class 1E Continuous Duty Motors for Nuclear Power Generating Stations," and member of Sub-Committee 2 of the Nuclear Power Engineering Committee.



TECHNICAL SESSION 3

Aging of Electrical Equipment

August 30, 1988

Session Chairman

DR. CHARLES E. ROSSI
Director, Division of Operational Events Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

LIFE ASSESSMENT FOR ELECTRICAL AND INSTRUMENTATION SYSTEMS AND EQUIPMENT - IEEE/NPEC WG 3.4 STATUS REPORT

S. Sonny Kasturi

Abstract Continued safe operation of the existing inventory of nuclear power plants during their current license terms and beyond requires a better understanding of the aging-induced degradation of the installed equipment and ability to project their expected life. The Codes and Standards community must respond to the needs of the industry in support of this. To serve this need, the Nuclear Power Engineering Committee of the IEEE initiated its efforts through Working Group 3.4. This paper presents a report on the progress of WG 3.4 since its inception two years ago, which led up to its present efforts on developing a guidelines document on life assessment of electrical and instrumentation equipment.

Introduction

Operation of the existing inventory of nuclear plants during the advanced stages of their current license terms and beyond demands a better insight into the in-service aging-induced degradation of the safety systems and equipment. Recognizing the need to support the related industry initiatives to assess remaining useful life of systems and equipment, the Nuclear Power Engineering Committee of IEEE established its Working Group 3.4 in December 1985. The objectives then were:

- To establish a resource group within NPEC to serve as the focal point for matters related to nuclear plant life extension (PLEX), and
- To be prepared to respond to the needs for standards revision and creation as the case may be to support this industry initiative.

At present, the WG has begun development of a guidelines document to address the related technical issues. This paper presents a report card with a view to sharing the knowledge gained.

Methodology:

The working group began its activities by addressing each of the following areas:

1. Technical considerations involved in estimating and extending life of electrical and instrumentation and control (I&C) equipment.
2. Adequacy of existing IEEE[1] ISA[2] & NEMA[3] standards to address life estimation and extension.
3. Use of surveillance, maintenance, and condition monitoring techniques as tools for estimating remaining life.
4. Review of ongoing national and international aging research efforts with respect to their applicability to addressing life extension.

This work was performed by 5 task action groups consisting of 4 to 6 members each. A discussion of the specific technical or programmatic areas examined and the conclusions thereof follows:

1. Identification and Discussion of Issues Related to Aging Degradation

The technical and programmatic issues related to age degradation of electrical and I&C equipment (hereafter referred to as ECI equipment) used in nuclear power plants were identified and categorized (see Table 1) as follows:

Category 1: Issues which are "codifiable," i.e., those issues which can be incorporated into the standards through standard or guide revisions or initiations.

Category 2: Issues which are technology limited and require additional research, development and demonstration (RD&D) in order to resolve the technical aspects. Hence, they must be addressed in other R&D forums.

Category 3: Issues which are policy-oriented, and hence, beyond the working group's scope and charter.

2. Equipment and Systems

For ECI equipment, life extension can be addressed by one of the following three methods:

1. Monitor - Ongoing monitoring and analysis of the aging-induced degradation and refurbishment.
2. Replace/Refurbish - Periodically replacing components and refurbishing equipment.

---

[1] IEEE - Institute of Electrical and Electronics Engineers.

[2] ISA - Instrument Society of America.

[3] NEMA - National Electrical Manufacturers Association.

3. Systematic Upgrade - Upgrading systems and equipment systematically to keep abreast of technology.

Table 1. Issue Listing and Categories

Issues	Category
1. Aging Degradation Mechanisms	
1.1 Insulation Aging	2
1.2 Electric Coil Life	1
1.3 Buried/Immersed Cable Problems	1
1.4 Shielded Cable Jacket Life	1
1.5 Degradation Process Identification	1,2
2. Inspection, Maintenance, Periodic Testing, and Surveillance	
2.1 Degradation Detection	1
2.2 High Voltage Testing	1
2.3 PRA Application	1
2.4 In Situ Testing Methods	2
3. Recertification of Qualification	
3.1 Equipment Requalification	1
3.2 Artificial Aging Techniques Re-evaluation	2
3.3 Operational Readiness Assessment	3
4. Data Collection, Evaluation, Storage, and Retrieval	
4.1 Records Management	1
4.2 Data and Analysis Requirements	1
4.3 Plant Experience Records	1
5. Life Assessment Evaluation	
5.1 Electronic Equipment Life	2
5.2 Remaining Service Life Calculations	2
5.3 Environmental Conditions/Controls	3
6. Replacement/Refurbishment	
6.1 Equipment Obsolescence	3
6.2 Use of Technology Upgrade	3
7. License Renewal	
7.1 Plant-Specific Licensing Technical Requirements	3
7.2 General Licensing Technical Requirements	1

A list of safety-related ECI equipment categories were first compiled. From this list, a representative sample was evaluated in detail to gain insight into which method of life extension would apply to which category of ECI equipment. Based on this review, a set of criteria for use in categorizing the equipment by its applicable life assessment techniques was developed. Table 2 provides a list of the samples examined and their LEX categories.

Table 2. Sample ECI Equipment Examined and their LEX Methods.

Equipment	LEX Method
Diesel Generator	1
Penetrations	1
Cables	1
Motors	2
Inverter/Charger	3
Switch Gear	1
Reactor Protection System	3

3. Status Of The Existing Standards

The adequacy of the existing IEEE Nuclear Standards (about 45) in addressing the identified aging degradation issues was then evaluated. About 20 of these standards which encompassed the sample group of equipment were subject to a detailed examination.

4. State Of The Technology

A quick-look survey was then performed of the ongoing and completed research and development projects related to detecting, monitoring and managing aging degradation. Using this information, age degradation issues listed and categorized earlier were assessed to identify those issues where technology development has reached a level to permit their treatment by the standards community.

5. Maintenance Activities Related To Age Management

Current utility maintenance, surveillance and test practices were examined to gain an understanding of their effectiveness in managing aging-induced degradation of electrical and I&C equipment and systems. This effort identified the need for additional guidelines for detecting and managing aging-induced degradation of ECI equipment.

Conclusions

Several technical and administrative issues need to be addressed to support nuclear plant life extension. Table 1 presents a listing and categorization of the technical and programmatic issues related to aging-induced degradation for electrical and I&C systems and equipment.

Existing NPEC sponsored standards do not contain any criteria or limiting requirements to inhibit life extension efforts. However, they are insufficient in providing a set of cohesive and consistent guidelines to support assessment of remaining life or extension of life of ECI equipment and systems. Guidance is needed on how to demonstrate continued qualification of qualified equipment.

Industry needs and, if available, could immediately use guidelines related to the criteria for assessing remaining life and providing a technical basis for extending life of ECI systems.

PLEX-related RD&D efforts are on-going in six of the seven ECI equipment and systems categories examined. The RD&D efforts appear to be well distributed in terms of the PLEX-related technical issues evaluated by the working group. New and continued RD&D should be encouraged in addressing electronic equipment and insulation system life assessment techniques, condition monitoring, data collection, and analysis techniques to identify and mitigate aging-related degradation.

In some cases, aging degradation detection and mitigation technology is not mature enough to permit its inclusion in the existing nuclear industry standards. However, a sufficient body of knowledge does exist to provide a technical basis for development of a guidance document for use by the industry in support of life assessment and extension for ECI equipment.

Status of Working Group Activities

The working group prepared an action plan report and presented its conclusions to the Nuclear Power Engineering Committee of IEEE, and other Peer Review Groups such as the Board of Nuclear Codes and Standards Steering Committee on PLEX, NUPLEX Codes and Standards Subcommittee. Further, the working group recommended that it be authorized to develop and issue a stand-alone guidelines document to address life assessment for safety-related ECI equipment. Its action plan report, which is under review by the members of NPEC, would be available for general distribution within three months.

The WG recommendation to develop and issue a stand-alone guideline document was concurred with by the various Peer Groups and accepted by NPEC in March 1988. The working group has begun development of this guidelines document. The goal is to get a document for trial use out by 1992, just in time to support the first license renewal application.

As chairman of the working group, the author of this paper requests the user and supplier community to participate and contribute to the expeditious development and issuance of the proposed guidance document.

Bibliography

1. "Interim Report on Plant Life Extension," WG 3.4 Report to NPEC.
2. L.C. Meyer. "Nuclear Plant Aging Research on Reactor Protection System," NUREG/CR-4740.
3. W. Gunther et al. "Detecting and Mitigating Battery Charger and Inverter Aging," NUREG/CR 5051.
4. J.P. Vora. "Nuclear Plant Aging Research (NPAR) Program Plan," NUREG 1144, Rev. 1 9/87.
5. SECY 87-179, NRC Policy Memo To The Commissioners. "Status of Staff Activities To Develop A License Renewal Policy."

UNDERSTANDING ELECTRICAL WIRING AGING
Analogies from U.S. Navy Experience

F.J. Campbell & A.M. Bruning

Introduction

The U.S. Navy is experiencing a maintenance problem with aircraft wiring that is caused by the use of an aromatic polyimide insulating material--Kapton[1]-- which is aging at a faster rate than was initially expected. Because the aging mechanism is a function of the moisture and heat in the ambient atmosphere, we can see possible analogies between the lessons learned by the U.S. Navy and those which are occurring in some of the wiring installed within nuclear power plant containments. We recognize there may be substantial differences in rates of aging, dimensions, environments, and degree of physical stress as compared to wiring on aircraft routinely making aircraft carrier landings. However, from our knowledge of power plant engineering, we are confident that analogies do exist. For example, information in reference (1) indicates a Kapton wiring failure in a power plant which is similar to those experienced in Naval aircraft.

Description of a Typical Failure Mode

Figure 1 illustrates a laboratory induced cracking failure of Kapton. This failure is a consequence of shortening of the polymer chain with a moderate operating temperature in a high humidity environment. As a consequence of the shortened polymer chain the material is no longer capable of supporting the strain of the bending of the wire, in this case 10%.

Figure 2 illustrates the drastic loss of mechanical properties effect on Kapton. A detail discussion of the Arrhenius technique used in this figure can be found in reference (2). For those not familiar with the Arrhenius relation, Figure 3 illustrates the log of time versus the inverse of absolute temperature relation which is applicable to many organic aging mechanisms. The dotted line in Figure 3 correlates the temperature with the inverse absolute temperature. It is common practice to submerge this functional step on standard graphs by showing only the temperature on the abscissa.

Having developed the Arrhenius relation from fundamental thermodynamics, including the interacting terms of moisture and temperature, we are in the process of including the energy of mechanical stress. To date our theoretical insights show correlation with laboratory and field data.

The life shortening process is illustrated by Figure 4, taken from reference (3). This illustrates the relation between the breaking strain on the ordinate versus the mean molecular weight of the Kapton polymer on the abscissa.

Figure 5 illustrates a generalized set of physical property hyperbolic characteristics which, with proper choice of B constant and elongation at infinite molecular weight, provide insight for Figure 2. For instance, if we are concerned with the strain capability of Kapton, using the B=0.1 curve, which has a strain withstand ability of 70% (at 1 pu) (reference 4) at a mean molecular weight of one unit, this curve indicates the ability to withstand strain will fall to 0 with about a 3% (.03 pu) reduction in mean molecular weight. There are indications that other physical properties of Kapton and other organic insulations follow this hyperbolic relation. Further, we believe that many aging characteristics of polymers arise from the progressive scissioning of the polymer chain due to the Arrhenius law, which relates reaction rate to temperature.

Multifactor Stress

The results in Figure 2 and Figure 3 are discussed in more detail in reference (5). To summarize these results, the presence of moisture causes a change in the rate at which the Kapton molecule scissions, which leads to a decrease in the ability of Kapton to withstand the strain arising from a bend in the Kapton insulated conductor. A major challenge in determining the aging rate of Kapton is to know the levels of moisture, strain, and temperature at which the insulation has been and will be subjected. Predicting the remaining life of nuclear power plant wiring requires understanding of these effects for each material.

Condition Monitoring

Although it is not within the scope of this paper to discuss "in situ" routine checking of a power plant wiring system, we would like to briefly review some methods and experiences in this area to broaden your thinking on this subject. Reference (6) details one controlled experiment with 220 aircraft. A simple 1000 v ac hipot test indicated 1359 insulation faults. This method cut in half the unexpected wiring failures. In addition to hipot and other similar techniques, time and frequency domain reflectometry, programmed artificial intelligence, and chemical sampling tests of existing wiring are all methods that may provide capability to assess the aging that has occurred and, thus, project the remaining life of the insulation.

Conclusion

The critical technical issue is centered on the ability to develop the theoretical formula for determining the deterioration rate from the combination of fundamental kinetics and thermodynamic principles needed to express the multifactor stress effects on the commonly utilized Arrhenius formula. This formula will then be utilized to determine the functional life of the wire in service using a computerized integration method to integrate deterioration rates based on the wiring's operating environment. Also, routine NDE testing will reduce the unexpected failures of wiring systems. The combination of these two could provide you the greatest amount of reliability assurance in nuclear power plant wiring.

References

(1) U.S. Nuclear Regulatory Commission, "Degraded Motor Leads in Limitorque DC Motor Operators," <u>IE Information Notice No. 87-08</u>, SSINS No. 6835, IN 87-08, February 4, 1987.

---

[1] Trademark of the Du Pont Co.

(2) Campbell, F.J. and Brancato, E.L., "Determination and Application of Thermal-Life Characteristics of Aerospace Wires," Insulation, October 1963.

(3) Wallach, M.L., "Structure-Property Relations of Polyimide Films," Journal of Polymer Science, 1968, Part A-2, Volume 6, pages 953-960.

(4) Du Pont Company, "Physical and Thermal Properties," Kapton Polyimide Film Summary of Properties, Technical Bulletin No. E-72087, January 1985.

(5) Campbell, F.J., "Temperature Dependence of Hydrolysis of Polyamide Wire Insulation," IEEE Transactions on Electrical Insulation, February 1985, Volume EI-20 No. 1.

(6) Wilhelm, H. and Mittag, H., "Electrical Insulation Test of Aircraft Wiring by Means of High-Voltage," LFM 321 Aircraft Electrics, Messerschmitt-Bolkow-Blohm GmbH Helicopter and Military Group, Manching Plant, April 28, 1986.

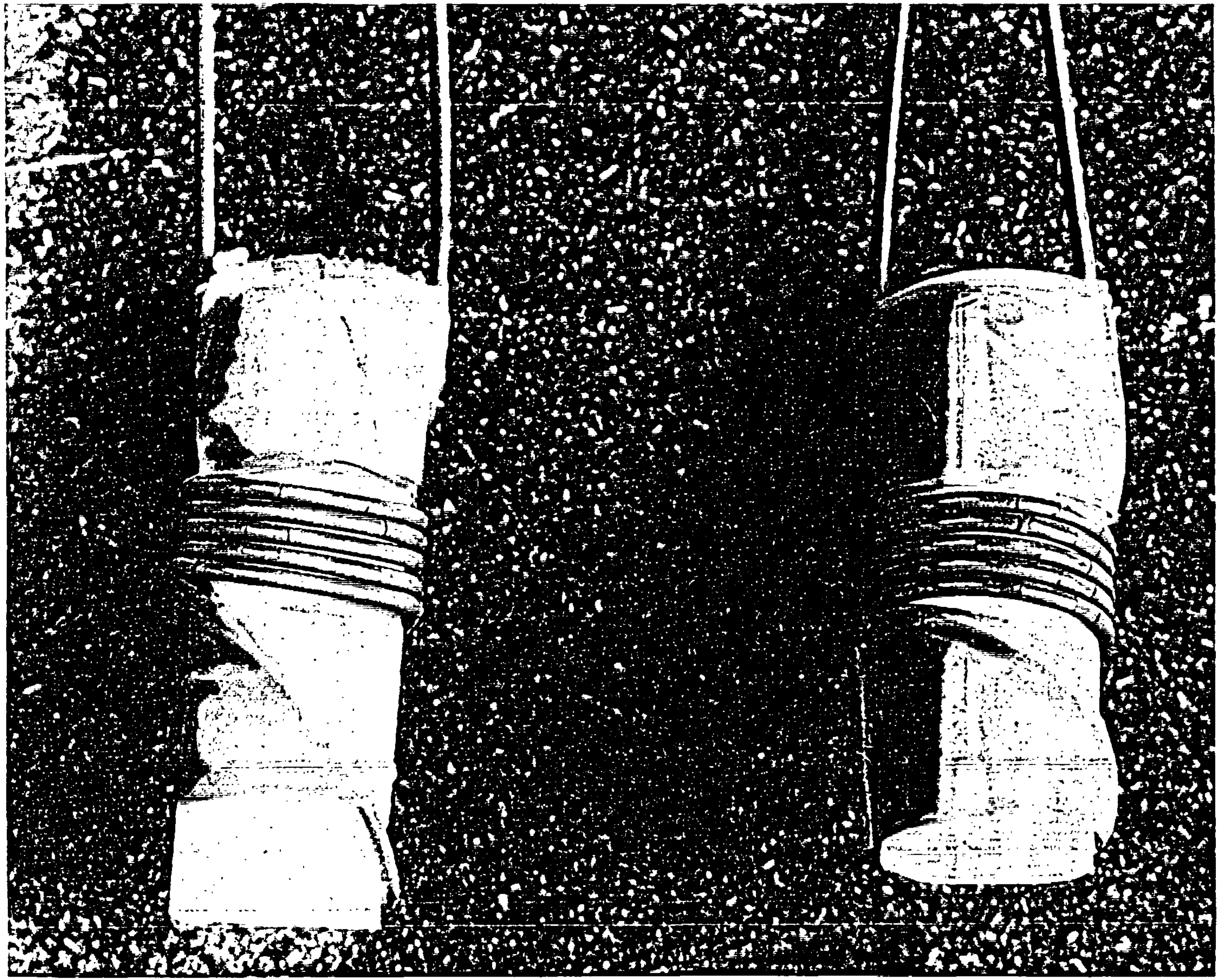


Figure 1

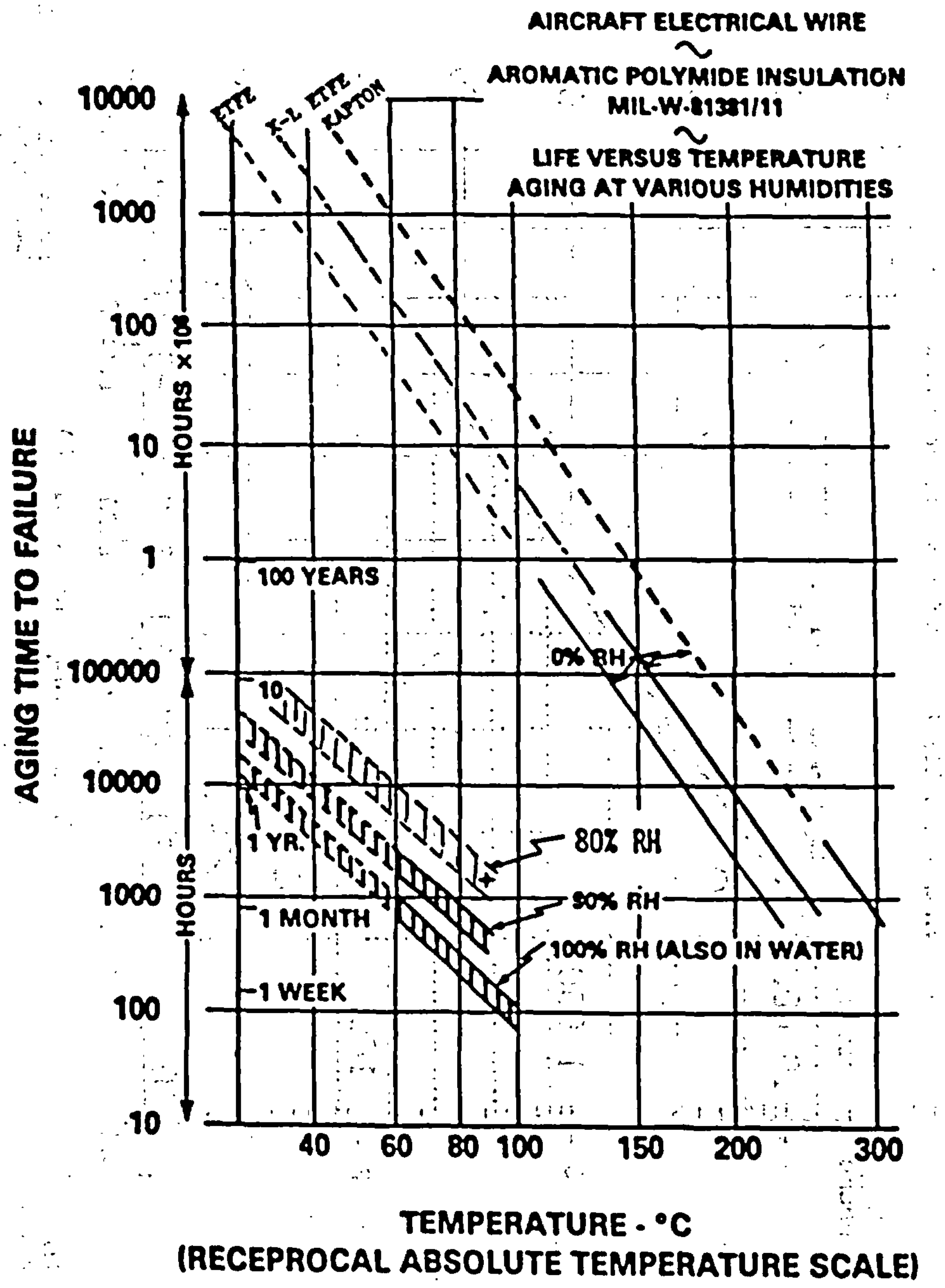
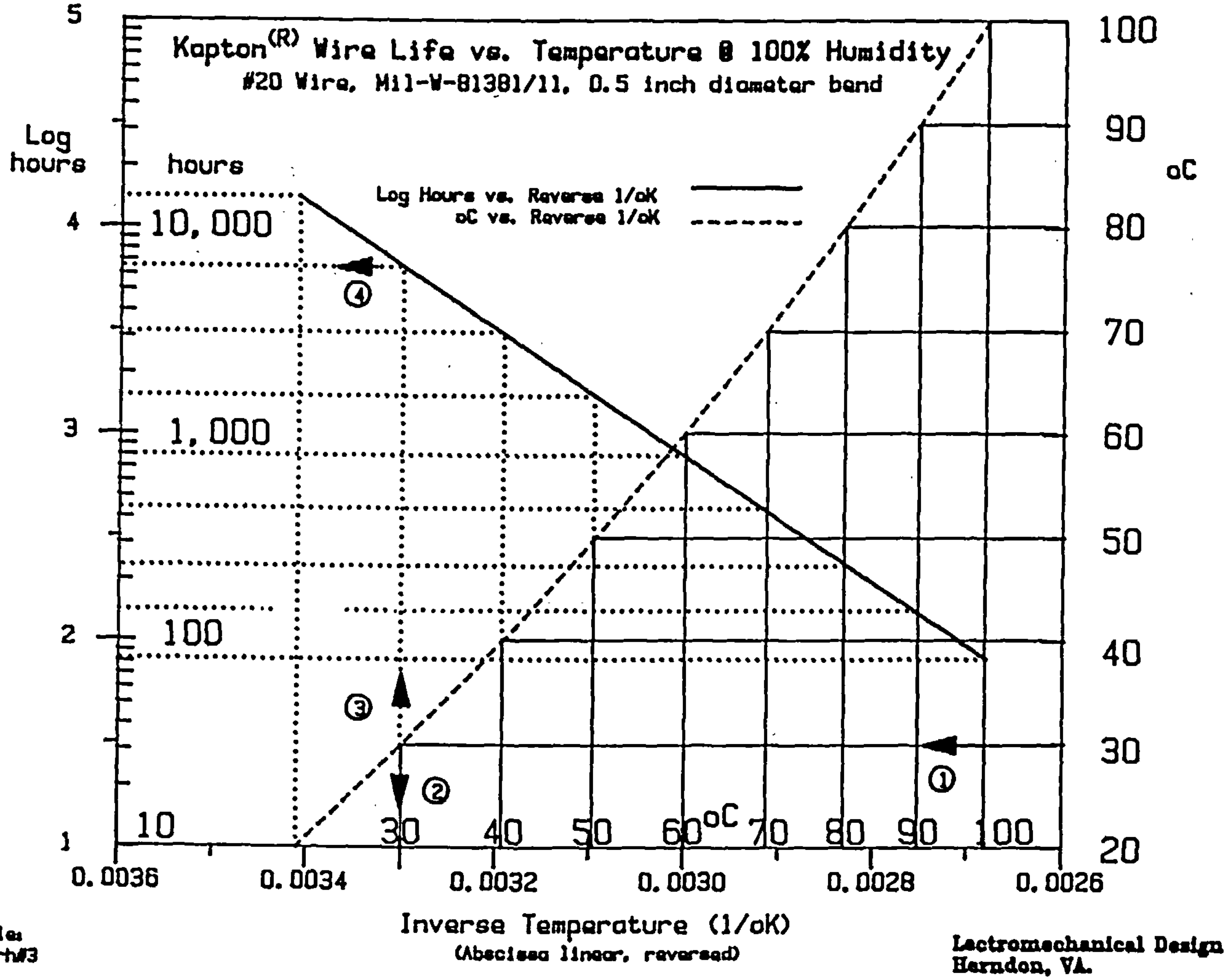

AIRCRAFT ELECTRICAL WIRE
AROMATIC POLYMIDE INSULATION
MIL-W-81381/11
LIFE VERSUS TEMPERATURE
AGING AT VARIOUS HUMIDITIES
ETFE
X-L ETFE
KAPTON
AGING TIME TO FAILURE
HOURS × 10⁵
HOURS
10000
1000
100
10
1
100000
10000
1000
100
10
100 YEARS
10
1 YR.
1 MONTH
1 WEEK
0% RH
80% RH
90% RH
100% RH (ALSO IN WATER)
40
60
80
100
150
200
300
TEMPERATURE - °C
(RECEPROCAL ABSOLUTE TEMPERATURE SCALE)

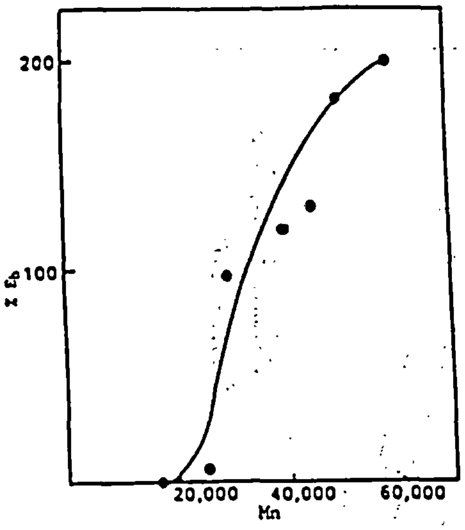
Figure 2



File:
Arrh#3

Lactromechanical Design Co.
Herndon, VA.

Figure 3



% E_b is the percent elongation (mechanical strain) at which the polyimide insulation fractures.

Mn. is the number-average molecular weight of the polyimide molecules in the chain.

Relationship between the elongation at break and the molecular weight of the polyimide

Figure 4

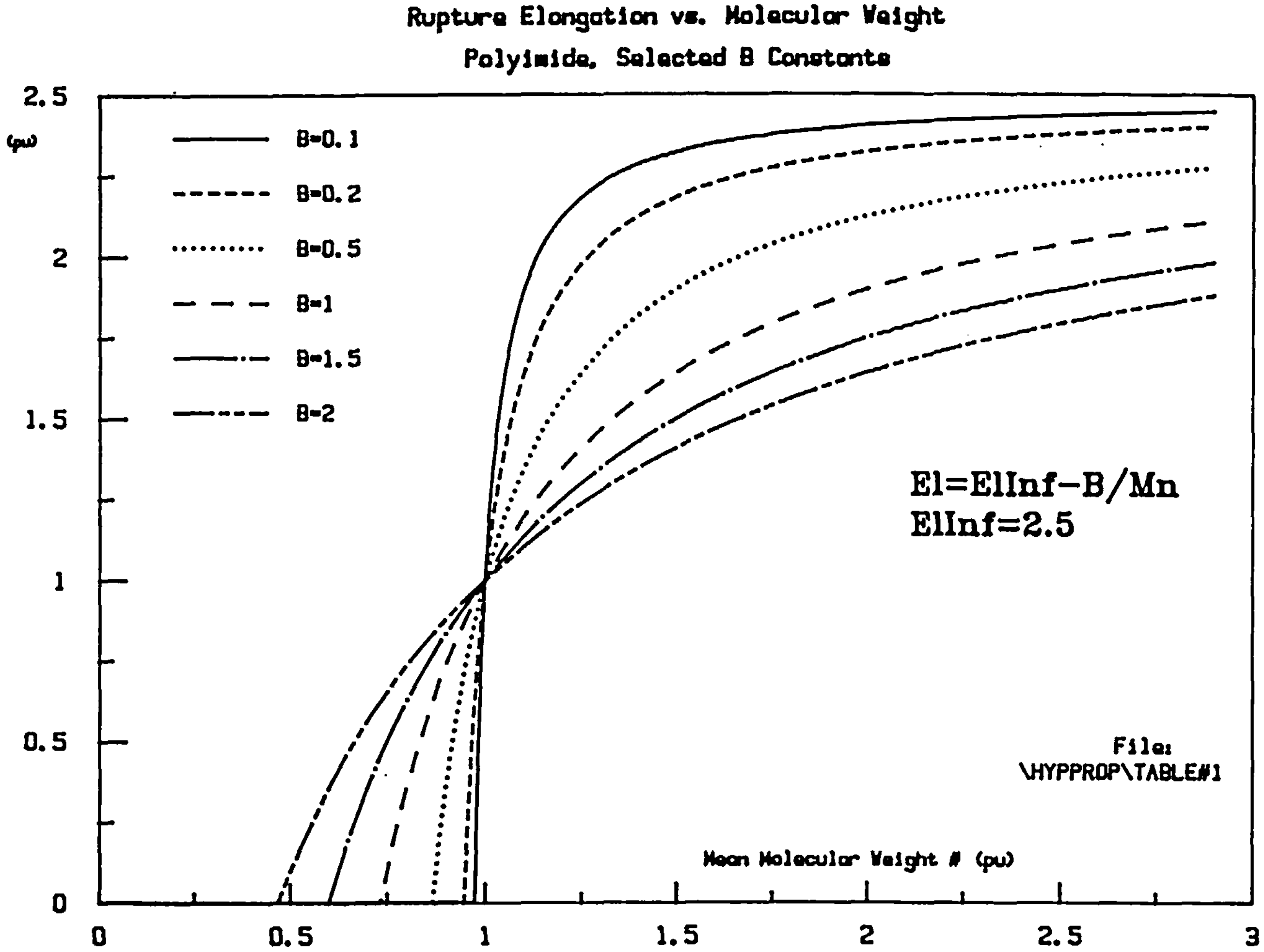

Rupture Elongation vs. Molecular Weight
Polyimide, Selected B Constants
2.5
(pu)
2
1.5
1
0.5
0
B=0.1
B=0.2
B=0.5
B=1
B=1.5
B=2
El=ElInf−B/Mn
ElInf=2.5
File:
\HYPPROP\TABLE#1
Mean Molecular Weight # (pu)
0
0.5
1
1.5
2
2.5
3

Figure 5

NEW METHODS FOR IN-SITU RESPONSE TIME TESTING OF PRESSURE SENSORS IN NUCLEAR POWER PLANTS

H.M. Hashemian
K.M. Petersen

ABSTRACT

Two methods are presented for response time testing of pressure sensors. These methods are called the "power interrupt" test and "noise analysis." The power interrupt test is applicable only to force balance type sensors but noise analysis is not generally restricted to any sensor type. The advantage of these methods is that they can be performed remotely while the plant is operating. These tests were developed to overcome the shortcomings of the conventional method. The conventional method is referred to as the "substitute process variable" test. This test is tedious, requires physical access to each sensor, and does not account for the effect of sensing line blockages and/or bubbles on the system response time.

INTRODUCTION

The response times of safety system pressure sensors in nuclear power plants are measured periodically to satisfy technical specifications and regulatory requirements. To date, more than ten years of testing has been completed in many plants involving about 40 to 80 sensors per plant. Unfortunately, a majority of these tests are believed to have produced erroneous results due to difficulties with the conventional test method and test procedures. Of course, the procedures can be improved but this may no longer be necessary because simpler test methods, as described in this paper, are now available which not only overcome the difficulties of the conventional test but also permit remote testing of pressure sensors while the plant is operating. These methods will reduce the burden on utilities and will provide useful results for assessment of aging effects on the dynamic response of pressure sensors. The term pressure sensor is used here as a general term to cover pressure, level, and flow sensors of the type used in light water reactors (LWRs).

HISTORICAL PERSPECTIVE

The concern about response time testing of safety system sensors in nuclear power plants was initiated in the mid 1970s. At that time, work was underway at the Oak Ridge National Laboratory (ORNL) to develop a method for in-situ response time testing of thermocouples for the Liquid Metal Fast Breeder Reactor project. This work served as the basis for a project at the University of Tennessee to develop in-situ testing capability for resistance temperature detectors (RTDs) which are used for safety system temperature measurements in nuclear power plants. This project was funded by the Electric Power Research Institute (EPRI). This project provided an exact method called the loop current step response technique for quantitative testing of RTDs.(1) During the same time period, EPRI sponsored two other projects for development of methods for response time testing of pressure sensors. One of those projects resulted in development of the "substitute process variable" test(2) and the other project provided information about the applicability of noise analysis for qualitative response time testing of temperature and pressure sensors(3) in general.

As a result of the EPRI projects, sensor response time testing capability became available in the late 1970s and has been used in nuclear power plants worldwide. As a result, it is now known that the concern about response time of RTDs has been justified as these sensors can suffer degradation as they age in a process. In the case of pressure sensors however, the effect of aging on response time is not known inspite of numerous tests which have been performed on thousands of pressure sensors in nuclear power plants. This is because the majority of the test results are believed to be erroneous due to inherent empirical difficulties with the conventional test method. These difficulties and a desire for in-situ testing capabilities led to new research sponsored by the NRC in the early 1980s at ORNL(4) and the University of Tennessee.(5) This research and work by others have provided the foundation for the new test methods now available.(6,7)

It must be mentioned that methods have also been developed for in-situ response time testing of neutron sensors.(8,9) However, routine response time testing of neutron sensors has not been performed in nuclear power plants.

REGULATIONS AND STANDARDS

The sensor response time testing question is addressed in several documents including the IEEE Standard 338, Regulatory Guide 1.118, ISA Standard 67.06, and NUREG-0809. The current consensus is that sensor response time testing should be done and that acceptable test methods are now available. In the past, it was not clear that the response time of the sensor itself must be included in determining the overall response time of a protection channel. Recent concern about potential degradation of sensors has indicated that the sensor should be included because it is a major contributor to the channel response time and is the component which is usually located in a harsh environment.

As a result of NRC enforcement of sensor response time testing requirements, the availability of suitable test methods, and some aging research currently underway, a reliable database is expected to become available to determine the effect of aging on response time of pressure sensors. In addition, EPRI intends to initiate a research project in the Fall of 1988 to identify if there are pressure sensors whose response time degradation may be detectable during static calibration.(10) The combination of the activities mentioned is expected to provide the information needed to identify those sensors which must be tested and to establish objective test frequencies and replacement schedules which may be needed due to aging degradation. Presently, pressure sensors are often tested at each refueling outage and there is no reliable information to determine if this testing interval is too short or too long.

RESPONSE TIME TESTING METHODS

Three methods are available for response time testing of pressure sensors. These are the substitute process variable test, the power interrupt test, and noise analysis. The power interrupt test and noise analysis are called in-situ methods, meaning that they can be performed on installed sensors while the process is operating. These methods are described below.

Substitute Process Variable Test

This method requires a pressure signal to be generated and applied to the pressure sensor and simultaneously to a high-speed reference sensor (Figure 1). A few plants use a step pressure signal but most tests employ a ramp input. This is because design basis accidents are usually defined in terms of pressure transients that approximate a ramp, and the only commercially available equipment for performing this type of test uses a ramp pressure signal.

The conventional test equipment generates a pressure ramp using controlled leakage of air from a high pressure to a low pressure cylinder (Figure 2). This gives an exponential pressure signal whose early portion approximates a linear ramp and is used for this test.

A new approach for the conventional method is to use a current-to-pressure (I/P) converter to generate the test signal. This is an effective approach for low pressure testing (up to about 100 psi). For higher pressures, the output of the I/P converter can be amplified and used. Pressure amplifiers that can provide suitable pressure outputs are not commercially available, but experimental prototypes have been developed and used successfully. Figure 3 is a simplified schematic of the new response time test equipment which uses an I/P converter and a pressure amplifier. The equipment is easier to use than the conventional hydraulic ramp generator and can produce a variety of test signals.

Power Interrupt Test

The power interrupt (PI) test is applicable only to force balance pressure sensors and has been validated for those manufactured by Foxboro Company. This is because Foxboro pressure sensors are the only force balance type sensors used in the safety systems of nuclear power plants.

The PI test is performed by a momentary interruption of the electrical supply power which is normally used to activate the sensor. The power to the sensor is simply turned off for a few seconds and then turned on. When the power is turned on, the sensor output is monitored and then analyzed to obtain its response time. It has been shown that the output of Foxboro force balance pressure sensors to a step change in pressure is the same as the output when the power is turned off and then on (Figure 4). The technique has been validated in the laboratory and successfully used in testing of pressure sensors in operating nuclear power plants.(4,7)

A typical PI test transient for a plant pressure sensor is shown in Figure 5. The good quality of test data is apparent in this figure. This transient is for a steam flow sensor tested with the PI method while the plant was operating at full power.

The equipment and procedure for performing a power interrupt test is simple and the testing of each sensor requires only a few minutes. The test does not interfere with plant operation and can be done at any time when the sensors are exposed to normal operating pressure. The data analysis is somewhat involved, but all the developmental work has been completed.(7)

Noise Analysis Technique

The noise analysis method is based on processing the natural fluctuations that exist at the output of a pressure sensor while the process is operating (Figure 6). This technique is not limited to a particular type of sensor as is the power interrupt test. Noise analysis is a completely passive test which does not require any perturbation of the sensor and is useful in many nuclear and non-nuclear applications for monitoring the performance of sensors and systems.

The technique relies on the assumption that a process sensor is normally exposed to random fluctuations (noise) which is inherent in most processes. Such fluctuations usually have a wide bandwidth and are often called white noise meaning that their spectrum is flat. It is simple to show that if a sensor is driven by white noise, its output power spectral density (PSD) can be used to estimate its frequency response. Therefore, in noise analysis, the procedure is to monitor the sensor output for a certain period of time and obtain its PSD. This is then used to determine the sensor transfer function from which the sensor response time is estimated. The data analysis can also be done in the time domain. An actual PSD plot from testing of a nuclear plant pressure sensor is shown in Figure 7. This data is for a steam generator level sensor in a pressurized water reactor.

It must be mentioned that response time testing of certain sensors, such as containment pressure sensors, is not practical with noise analysis. This is because the frequency and amplitude of containment pressure fluctuations are not usually adequate to sufficiently excite the sensor. In addition, the accuracy of noise analysis results will always depend on the validity of the assumptions made about the characteristics of the process fluctuations. If these characteristics are not suitable, the application of noise analysis must be limited to identification of changes in sensor response behavior rather than quantitative response time information. That is, noise analysis can still be used but only for monitoring for response time degradation from a reference value. Research is currently underway to determine the typical characteristics of process fluctuations in LWRs for testing of pressure sensors. This research is expected to provide information on the validity of noise analysis for quantitative response time testing of pressure sensors and associated sensing line systems.

A significant advantage of noise analysis over the other methods is that it can reveal gross blockages in the sensing lines. This is further discussed in the following section.

SENSING LINE EFFECTS

The sensing lines which bring the pressure information from the process to the sensors can have a significant influence on the overall pressure sensing system response time. The sensing lines are about 20 to 200 feet long with a majority less than 100 feet. Isolation valves in the sensing lines can fail, be inadvertently left partially or totally closed, or improperly lined up. In addition, the sensing lines can be blocked by contaminants, contain voids, or even freeze. All these events can cause improper system response which may be detected by analysis of noise data from the affected sensors. It is important to point out that continuous monitoring of pressure sensing systems with noise analysis is the only practical way to monitor for sensing lines that may become totally blocked.

COMPLICATIONS AND REMEDIES

A number of complications may be encountered in testing pressure sensors. These complications are due to sensor characteristics and have different effects on different test methods. Some examples are discussed below.

Oscillatory Sensors

The response of some pressure sensors to a step or ramp input can have an oscillatory component. Figure 8 illustrates the response of an oscillatory sensor to a ramp input. These sensors are said to have underdamped transient characteristics. The oscillation depends on the damping ratio of the sensor and the ramp rate. These oscillations create difficulty in calculating the sensor response time. As shown in Figure 8, the response time must be evaluated after the oscillations have died out. Otherwise, the results will depend on the time at which it is evaluated. The problem is that current test procedures require that the response time be evaluated at a specified setpoint which can fall in the oscillatory region and produce erroneous and nonreproducible results. This can even produce negative response time results if the oscillations are large enough to overshoot the input. Unfortunately, practical considerations usually prevent the test personnel from being able to observe this problem while performing the tests. Therefore, they can become confused as to why the results are different in successive tests or why negative values are sometimes obtained. The cure for this problem is to design the test to allow enough time for the oscillations to die out. There are two ways to accomplish this: start the test from an initial pressure that is further from the setpoint or use a slower ramp. Both of these approaches create additional questions which must be answered.

Starting the test at a pressure further from the setpoint raises a question about linearity of the input ramp. The hydraulic ramp generator, which is used in the conventional test, actually generates an exponential pressure signal. The procedures currently used were developed to obtain a pressure variation which is nearly linear over the duration of the test. Unfortunately, this may conflict with concerns about decay of oscillations in the sensor being tested. If initial pressures are set further from the setpoint, the tester should examine the pressure transient to confirm that adequate linearity is achieved.

Using a slow ramp to ensure decay of oscillations may prevent meeting design basis accident ramp rates in the tests. This probably will not affect the result, but it should be checked experimentally. Experimental work is underway to address this question.

Sensor Nonlinearities

Experience has shown that nonlinearities exist in some pressure sensors. As a result the sensor may respond differently to small perturbations than to large ones. This behavior may have a significant effect on noise analysis results depending on the level of sensor nonlinearity characteristics. Noise analysis techniques identify sensor response characteristics based on the sensor's response to small process fluctuations. It may not be valid to assume that these sensors would respond in the same manner to the large transients of interest for reactor safety. The significance of this limitation should be addressed before noise analysis is used for testing of these nuclear plant pressure sensors. Fortunately, there are only four types of pressure sensors commonly used in the safety systems of most LWRs. These are manufactured by Rosemount, Barton, Foxboro and Veritrak (or Tobar). Therefore, only a small effort is involved in establishing the testability characteristics of these sensors. The nonlinearity question of these sensors will be answered in a research project currently underway on validation of noise analysis for quantitative response time testing of nuclear plant pressure sensors.

Effect of Measurement Tolerance on Time Tolerance

Another concern in response time testing of pressure sensors is the effect of measurement tolerance induced by hysteresis and other effects in pressure sensors. All sensors have reproducibility tolerances which affect the uncertainty of their pressure indications. The reproducibility tolerance translates to a time tolerance in the response of the sensor to a ramp test signal. The time tolerance will depend on ramp rate as follows (Figure 9):

$$\text{Response Time Tolerance } (\pm) = \frac{\text{Reproducibility Tolerance } (\pm\ \%\text{ of span})}{\text{Ramp Rate } (\%\text{ of span/sec.})}$$

Typical reproducibility tolerance for nuclear grade pressure sensors is about 0.1 to 0.3 percent of span. For a slow ramp, the response time tolerance can be large. For example, in a plant procedure, the ramp rate specified for response time testing was 0.2% of span per second for a sensor with an allowable response time of 0.4 seconds. The uncertainty in the response time measurement in this case would be ± 1.0 second. This is an unacceptable measurement tolerance for a response time requirement of 0.4 second.

The solution to this problem is to avoid using slow ramps. This is allowed in the ISA standard 67.06 under the category, "alternate input pressure perturbance." However, experimental work is needed to determine optimum ramp rates that can minimize the tolerance problem, but are slow enough to comply with the ramp rates specified in the plant technical specifications.

CONCLUSIONS

The response time of pressure sensors is conventionally measured using a method called the substitute process variable test. This method requires a pressure test signal to be applied to the sensor and simultaneously to a high-speed reference sensor. The delay of the sensor with respect to the reference sensor is called the response time. A disadvantage of this method is that it requires physical access to the sensor. Therefore, new methods were developed to permit remote testing of sensors as installed in an operating process. The new methods are called power interrupt test and noise analysis. The power interrupt test is applicable to force balance type pressure sensors. It involves analysis of the sensor output after a momentary interruption of electric supply power which is normally used to activate the sensor. Laboratory validation work has shown that this output is equivalent to the response of the sensor to a step change in pressure. The noise analysis technique is based on monitoring the sensor output while the process is operating. The output often contains random fluctuations that can be analyzed to give the sensor transfer function from which the sensor response time is deduced. An important advantage of noise analysis results is that they include the effect of significant blockages and/or voids that may exist in the sensing lines.

REFERENCES

1. Hashemian, H. M.,Kerlin, T. W., Petersen, K. M., "New Methods for Response Time Testing of Temperature and Pressure Sensors," Proceedings of the International Symposium on Pressure and Temperature Measurment, ASME Winter AnnualMeeting, FED-Vol.4/HTD-Vol.58, pp 70-85, December 1986.

2. Foster, C. G., et al., "Sensor Response Time Verification," Report No. NP-267, Electric Power Research Institute, Palo Alto, CA, October 1976.

3. Currie, R. L., Mayo, C. W., Steven, D. M., "ARMA Sensor Response Time Analysis," Report No. NP-1166, Electric Power Research Institute, Palo Alto, CA, May 1980.

4. Buchanan, M. E., et al., "Measurement of Response Time and Detection of Degradation in Pressure Sensor/Sensing Line Systems," Nuclear Engineering and Design, No. 89, pp 91-99, May 1985.

5. Soares, A.J., "Study and Dynamic Modeling of Pressure Transducer that is Based on the Principle of Force Balance," Ph.D. Dissertation, The University of Tennessee, Knoxville, TN, December 1982.

6. Hashemian, H.M., Thie, J.A., Upadhyaya, B.R., "Sensor Response Time Monitoring Using Noise Analysis," Symposium on Reactor Noise, SMORN V Conference, Munich, West Germany, October 1987.

7. Analysis and Measurement Services Corporation, "A New Method for In-Situ Response Time Testing of Force-Balance Pressure Transmitters in Nuclear Power Plants," Topical Report, AMS Report No. TPI8801R0, September 1988.

8. Aldemir, T., Miller, D. N., "The Availability of Neutron Channels and Power Range Monitoring Systems with In-Situ Detection of Channel Degradation," Nuclear Technology, Vol.74, pp 267-271, September 1986.

9. Aldemir, T., Arndst, S. A., Miller, D. W., "Simulation of the Transient Response of Ionization Chamber to Bias Voltage Perturbations," Nuclear Technology, Vol. 76, pp 248-259, February 1987.

10. Presentation by Joe Weiss of EPRI at 1988 Informal Meeting on Reactor Diagnostics, June 8-10, 1988, Orlando, FL.

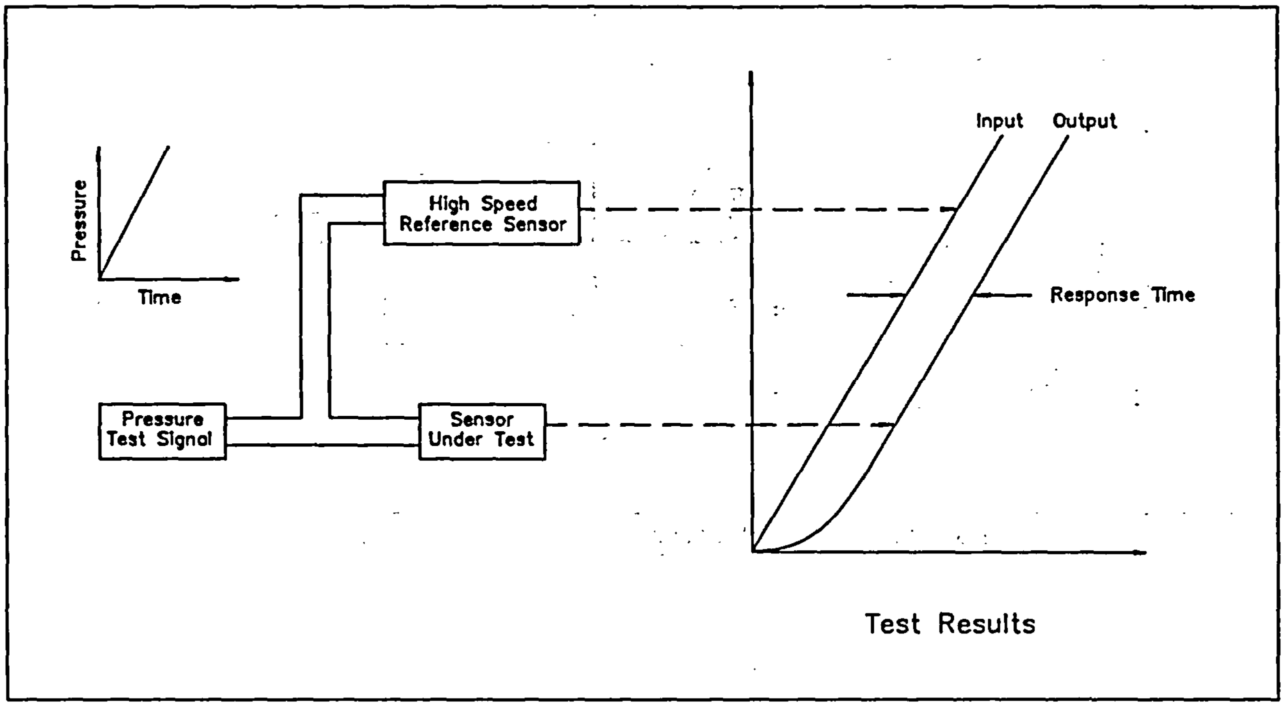


Figure 1. Pressure sensor response time test setup.

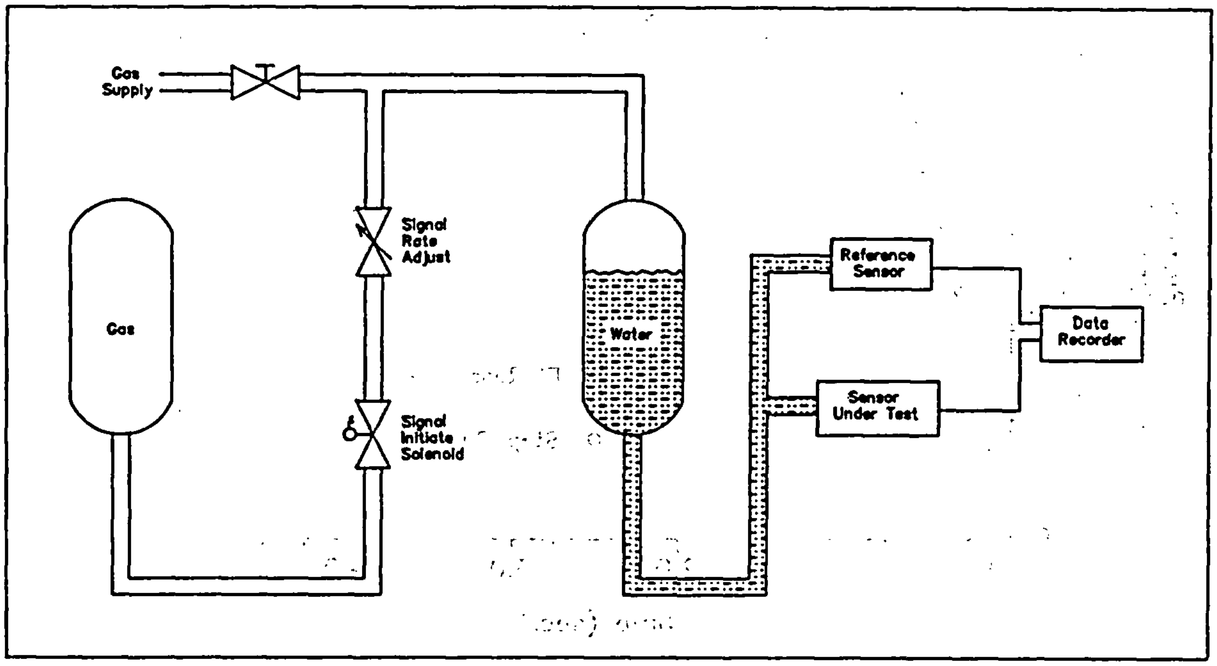


Figure 2. Simplified schematic of the conventional test equipment.

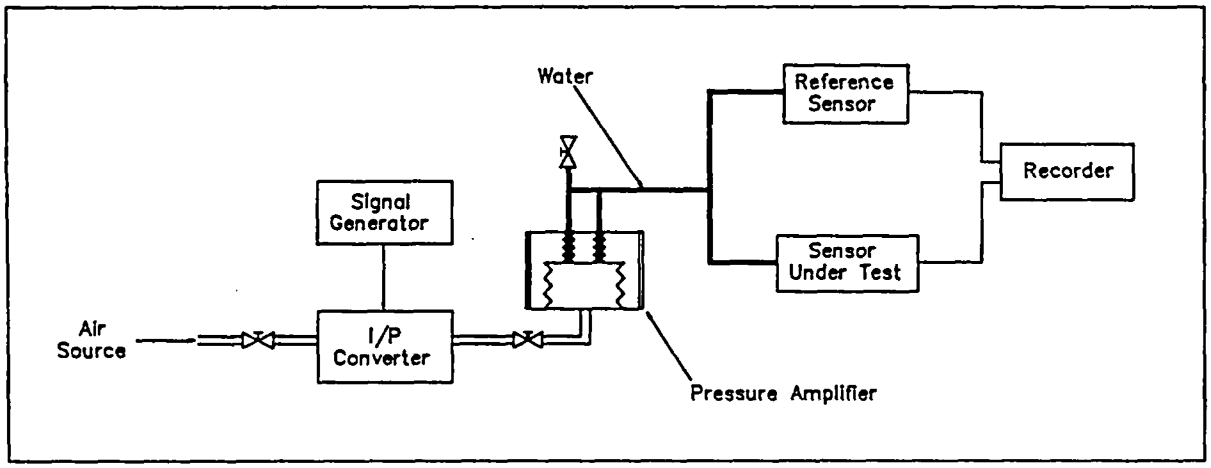


Figure 3. Simplified schematic of new test equipment.

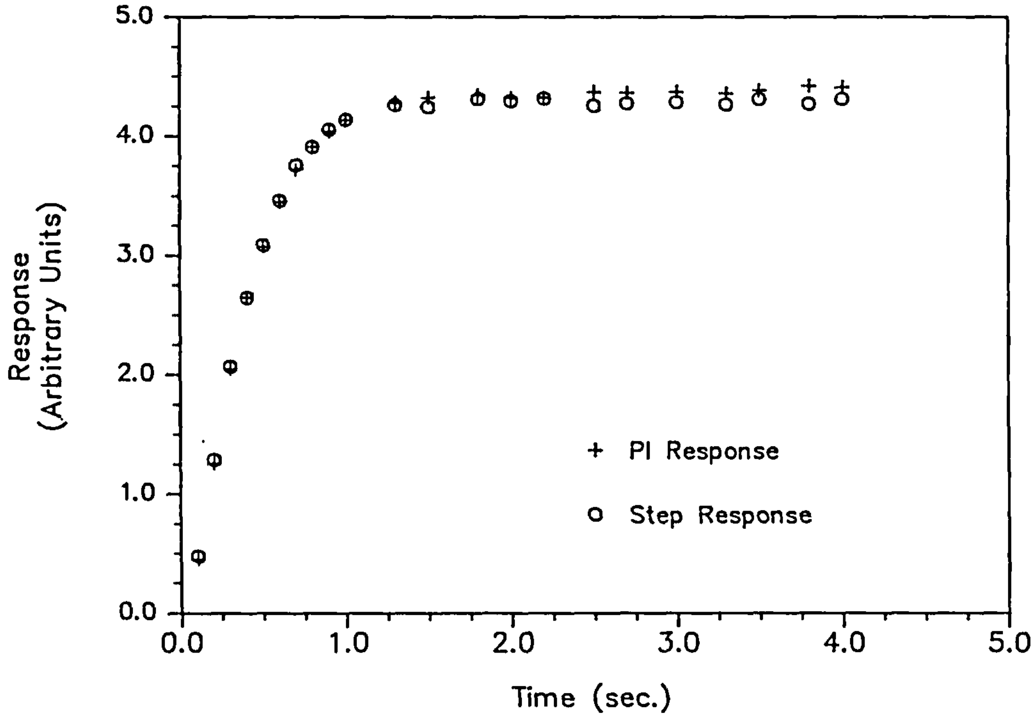


Figure 4. Response of a force balance sensor to a step pressure input and to PI test.

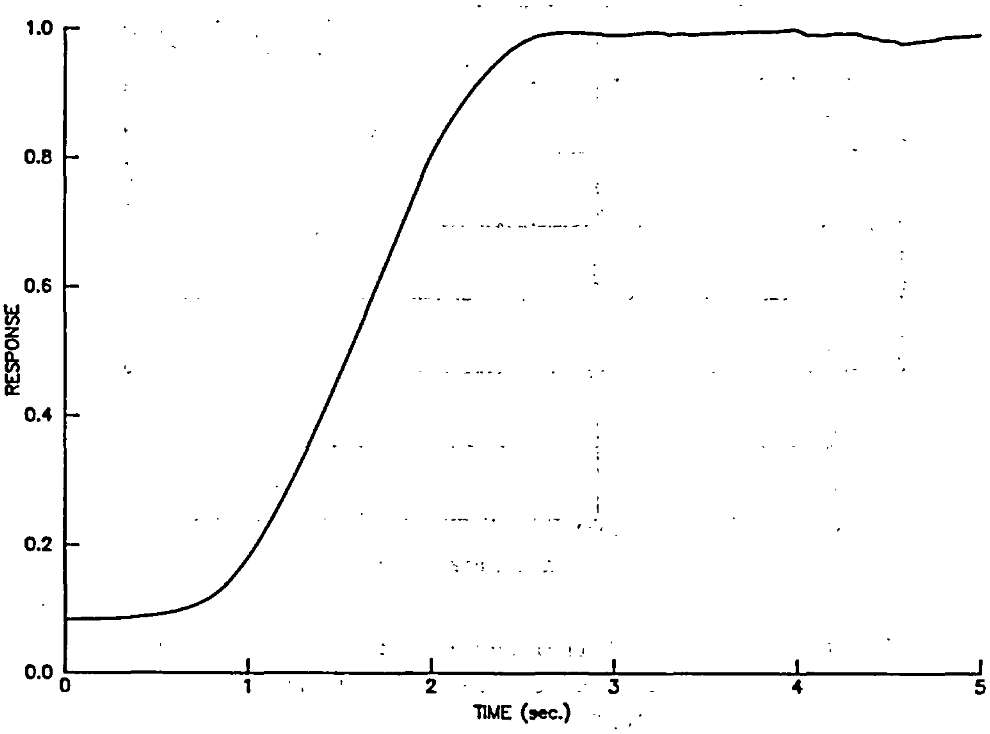


Figure 5. A typical PI test transient from a test at an operating nuclear power plant.

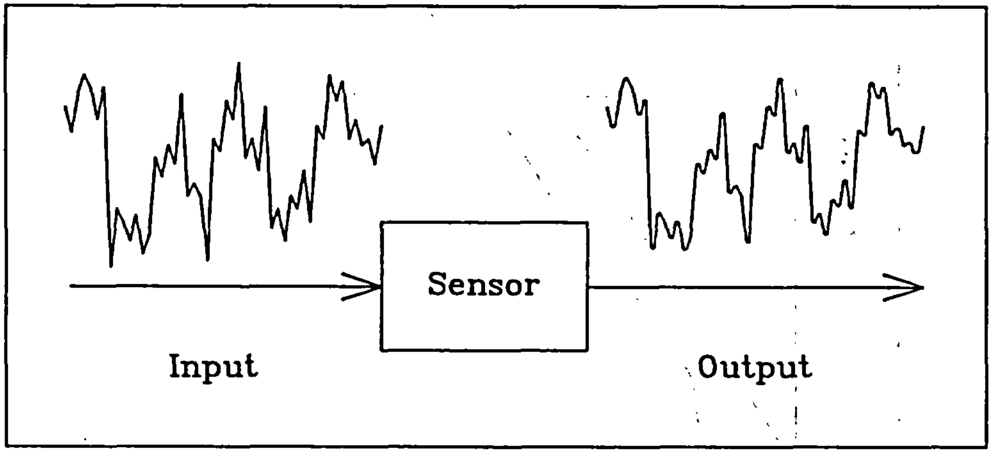


Figure 6. Illustration of noise analysis principle.

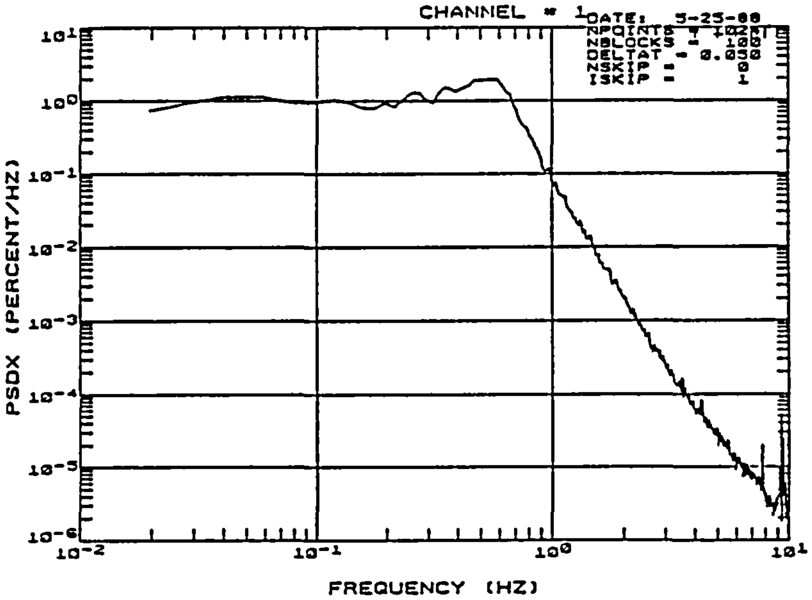


Figure 7. Power spectral density of a nuclear plant pressure sensor tested at normal operating conditions.

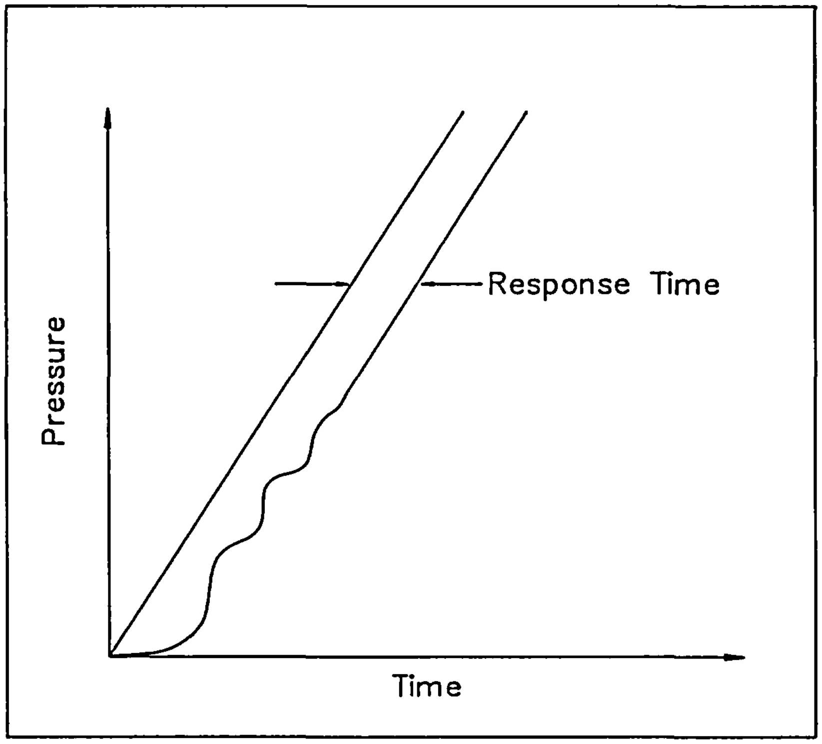


Figure 8. Illustration of ramp response of an oscillatory sensor.

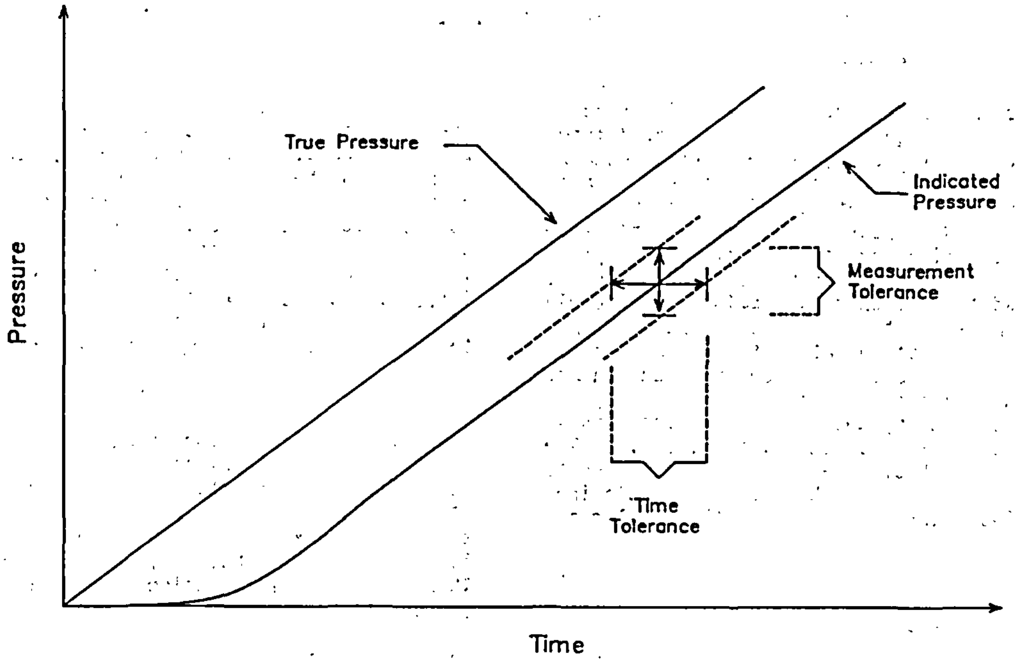


Figure 9. Illustration of effect of measurement tolerance on time tolerance.

THE EFFECTS OF A SEISMIC EVENT
ON
AGED MICROPROCESSOR AND ASSOCIATED INTEGRATED CIRCUITS
USED IN NUCLEAR POWER PLANT
CLASS 1E EQUIPMENT

J. Hicks and R. H. Jabs

Summary

This paper presents the results of a testing program conducted on microprocessors and associated integrated circuits including memories, data conversion, and high-speed logic components. These components are representative of those used in microprocessor-based post accident monitoring systems used in mild environments of nuclear power plants. The program was conducted to determine if any age degradation mechanisms exist that could result in the non-operation of these components when subjected to a seismic event at the end of a five-year operational period. Component selection was made with vendor and component technology being the major variables. Sixty-two samples representing 19 vendors were aged and seismically tested. The results were favorable. A few of the components were inoperable following the age preconditioning. A failure analysis was performed and the results are discussed.

Introduction

The requirement that aging be included as part of the environmental qualification test sequence for Class 1E equipment originated in IEEE Standard 323-1974. This requirement, often perceived as beyond the state of the art, has stimulated much industry discussion and debate, and extensive aging studies, particularly to address equipment located in mild environments in which the single design basis event (DBE) is the seismic event. The current nuclear industry thinking is reflected in the IEEE 323-1983 revision which states that aging be addressed "if the equipment is determined to have a significant aging mechanism." A significant aging mechanism causes "degradation during the installed life...that progressively and appreciably renders the equipment vulnerable to perform its safety function under DBE conditions."

The objective of this test program was to determine if significant aging mechanisms existed that could adversely affect the performance of microprocessor and supporting integrated circuits during a seismic event.

These components are used extensively both in the new generation of "smart instrumentation" added to nuclear power plants to meet the post accident requirements of Regulatory Guide 1.97, and in the new generation of plant protection systems.

The seismic sensitivity of integrated circuits has been studied in previous EPRI[1] and Westinghouse programs.[2] These programs, however, have evaluated the generation of integrated circuit technology used in Class 1E equipment designed in the early 1970s. These integrated circuits are small- or medium-scale integrated devices with typically 10 to 100 transistors in a package. The technology used was bipolar digital (Figure 1) and linear (analog) with a cell structure of about 50 to 100 square microns. The microprocessor and associated integrated circuits of this study were typically MOS technology (Figure 2) with cell size of 1-2 square microns and up to 100,000 transistors in a package.

Also emerging with this new technology were new integrated circuit vendors, particularly overseas vendors. So, because of the significant changes in the integrated circuit technology and the popularity of new vendors, it was deemed necessary to conduct this aging/seismic test program to determine if significant aging mechanisms existed that could adversely affect the performance of the integrated circuits during or after a seismic event.

The simulated operating period applied prior to the seismic testing was five years.

Degradation Mechanisms

The semiconductor industry has been and remains actively involved in conducting aging evaluations on its products. These tests are reliability studies. Although they do not address seismic sensitivity, they do determine the different degradation mechanisms that may affect these products. This detailed information is available from the vendors.[3,4]

The essential parts of an integrated circuit consist of the die (actual electronics), the bonds wires, the leads and the package as shown in Figure 3. Aging mechanisms can occur in any of these parts. Table 1 lists several of the failure mechanisms observed in integrated circuits and their relative frequency of occurrence. Such mechanisms are classified as electrical, chemical, or mechanical. The activation energy of the reactions is also shown where applicable. Activation energy is used with the Arrhenius relationship to determine the rate of the reaction at various temperatures. The Arrhenius model will be discussed later in more detail.

In general, electrical failure mechanisms occur when charged bodies are redistributed under the influence of the electric field present and any structural irregularities. The resulting concentration of ions or electrons causes degradation in the electrical performance of the device, which will eventually lead to inoperability. These types of failures are important in any evaluation of semiconductor reliability, but they will not be affected by a seismic event.

Chemical failure mechanisms, in general, are chemical reactions that degrade the required properties of materials or material interfaces in the integrated circuit in a way that leads to component failure. The most common chemical failure mechanisms are corrosion, intermetallic formation, and electromigration. These mechanisms result in failures when the material or interface affected can no longer perform its required function. In many cases, these are triggered by an electrical, thermal, or mechanical stress to the degraded material. For example, intermetallic formation may severely degrade the attach strength of a bond wire to bond pad but will not result in immediate inoperability. Repeated power cycling would have to occur before the contact is broken and the device becomes inoperative. Most chemical failure mechanisms will weaken materials and material interfaces in this way, thus increasing the probability of loss of function during a seismic event.

Mechanical failure mechanisms, in general, are those induced by physical stresses on the integrated circuit subcomponents. The physical stress can be due to direct shock to the component. This can be caused by external vibration or by rapid changes in temperature. The most common source of mechanical stress arises from the difference in thermal expansion properties between adjacent materials. Each time the temperature changes because of external conditions or power cycling, every material interface in the circuit is stressed to some extent. The industry has given much attention to minimizing these stresses.

Component inoperability induced by a seismic event will be mechanical, caused by the vibration induced stresses at one of the material interfaces. As noted, this is more likely to occur for a given vibration level if degradation of the interface has been precipitated by one of the chemical failure mechanisms previously mentioned.

Considering all the different aging mechanisms and their relative frequency, a 0.5 eV activation energy was chosen for this program. Selecting one activation energy to represent the reliability of a technology is consistent with industry practice. The activation energy of 0.5 eV is conservative and would allow for manifestation of any degradation induced by corrosion, which is most likely the most significant aging mechanism that could adversely affect the seismic performance of these components.

Further discussion on the details of various degradation mechanisms is given in the appendix.

Program Description

Component Selection

The microprocessor and associated integrated circuits tested in this program were representative of those used in Westinghouse-supplied microprocessor based Class 1E equipment. It was first necessary to identify, by vendor, all integrated circuits whose technology was significantly different from those digital and linear devices tested in early Westinghouse aging/seismic programs. This process included obtaining assembly drawings and approved vendors lists from OEM board manufacturers used in the Class 1E equipment, physically inspecting OEM production boards at the equipment manufacturer, and obtaining bills of material and approved vendor information for those printed circuit boards designed and produced by the vendor of the Class 1E equipment.

This information was provided to the engineers at the Westinghouse Semiconductor Control Center (SCC) who then recommended which integrated circuits should be chosen to be representative of the new technologies and vendors being used. The SCC was utilized because of its extensive expertise in integrated circuit technology, its state-of-the-art testing capability, and its established association with semiconductor vendors.

The microprocessors and associated integrated circuits selected were representative of the technologies (component families) and the vendors identified. Although there are many degradation mechanisms that may exist in a technology, none are sensitive to pattern metallization, that is, the arrangement of transistors to perform a particular function (counter, buffer, etc.). Pattern insensitivity means, for example, that the performance of a vendor's Bipolar Schottky family of integrated circuits can be established by testing one or a few representative components. Seismic performance is therefore independent of this one major variable which produces a wide range of electronic functions within a component family.

The same rationale extends to integrated circuits with larger metallization layouts such as a vendor's line of microprocessor and supporting integrated circuits (timers, programmable ports, communication interfaces). Manufacturing of these large-scale integrated devices in a very competitive market forces a vendor to invest heavily to provide high yields that are trouble-free. He then applies this manufacturing process as widely as possible to maximize the return on his investment.

Since the pattern metallization process is standard within a vendor's component family, the single variable in any family is the integrated circuit vendor. Each vendor's manufacturing process is typically different. The process is usually proprietary to his organization and is not shared without commercial agreements.

The component families tested and the vendors represented in this test program are given in Table 2. Sixty-two samples representing 19 vendors were tested. Of these samples, several vintage components representing the state of the technology in the early 1980s were included.

Baseline Testing

After representative components had been received, they were inspected for vendor markings and for any observable damage that may have occurred during shipping and handling. Each integrated circuit was then electrically characterized using automated test equipment such as the Gen Rad linear and digital testers and the Fairchild Sentry IV. A detailed characterization analogous to a good commercial dedication program verified that good components were being tested and provided data for comparison to post aging and post seismic testing data to be taken later.

The baseline testing also programmed each non-volatile memory (for example, EPROMS and NVRAMS) with a bit pattern. After programming, the EPROM windows were covered to avoid accidental exposure to ultraviolet light which could alter the stored pattern. Westinghouse uses this practice for all its Class 1E systems.

Aging

After completion of the baseline testing, typically six identical components of each sample were subjected to accelerated aging. The Arrhenius model was used to determine accelerated aging times for each component sample. This model relates an accelerated time to real time as follows:

$$ts = ta \exp\left[(-E/K)\left(\frac{1}{Ta} - \frac{1}{Ts}\right)\right]$$

ts = Service Life (qualified life)
ta = Aging Time
E = Activation Energy (eV)
K = Boltzman's Constant (8.617×10^{-5} ev/degrees K)
Ta = Aging Temperature (degrees K)
Ts = Service Temperature (degrees K)

A service temperature of 60°C and an activation energy of 0.5 eV were used as the primary parameters in this equation. The 60°C temperature accounts for component joulean heating and for the heat rise within a typical equipment cabinet. The 0.5 eV activation energy is conservative and allows manifestation of degradation mechanisms that have been found in reliability studies discussed earlier.

Aging temperatures were either 145°C, 130°C, or 105°C, depending on the manufacturer's limitation placed on his product. Aging times using the above parameters were 1272, 2136, and 5510 hours, respectively, to simulate a five-year operational period. During aging, all component leads were connected to prohibit damage from electrostatic discharge.

Following the thermal aging, baseline measurements were repeated on each aged component. Three components of each component sample that remained within manufacturer's specifications were then prepared for seismic testing.

Seismic Testing

Each integrated circuit was mounted on a printed circuit board (Figure 4), which was then installed in a card cage to simulate an actual in-plant installation. The card cage was mounted to a rigid test fixture that was bolted to the seismic stand.

Each integrated circuit was energized with a circuit that monitored its performance. For example, microprocessors were programmed to provide an output function, and the performance of memory integrated circuits was verified through read/write cycles from an auxiliary microcomputer. For some of the basic logic gates, a chart recorder monitored the performance of each gate as it was pulsed on and off.

The seismic testing was conducted in accordance with IEEE Standard 344-1975 using a multi-frequency input. The Required Response Spectrum (RRS) was obtained from the in-equipment response spectrum from the systems seismic qualification on all the microprocessor-based Westinghouse Class 1E equipment. The resultant RRS was broad band with an acceleration peak of 60g for the Safe Shutdown Earthquake in the control direction of the seismic stand.

Results

Aging and Seismic

The results of the age preconditioning showed that all component samples were not limited to a five-year operational period. Only four integrated circuits (about 1%) were inoperable or beyond manufacturer's specifications. These were considered random failures.

All aged integrated circuits that were seismically tested demonstrated acceptable results. No component-related intermittencies, shorts, or open circuits or loss of function was recorded during the seismic testing. Baseline testing was again performed following the seismic testing. Three components were found to be inoperable. A failure analysis was performed on these components.

Failure Analysis

Table 3 lists the components that were inoperable or altered during this test program. Using standard semiconductor industry techniques, they were physically dissected with the objective of identifying any cause-effect relationship between the aging/seismic stress and the electrical symptoms. The most frequently observed disorder was physical damage of the dies from either electrical overstress or voltage current overload. Three components that passed seismic testing were found not fully operational in the post-seismic baseline testing. This was attributed to handling oversights and to procedural mishaps during stress application. It is here noted that these integrated circuits are very sensitive to electrostatic damage and must be handled appropriately.

The EEPROM (Item 6, Table 3) continued to function within specifications. However, it exhibited a change in the pattern of ones and zeros that had been stored before application of the aging/seismic stress. This amounts to a change in the stored charge, most likely due to a software error during this program. During the physical examination, a 40 micron-diameter spot of titanium metal adhering to the semiconductor was found. It is believed to have been present from the day of manufacture.

Summary, Conclusions and Recommendations

A test program was conducted on a comprehensive sample of five-year-aged microprocessor and supporting integrated circuits to determine if any degradation mechanisms existed that could affect their performance during a seismic event. These integrated circuits were representative of those used in Westinghouse microprocessor-based Class 1E systems. Sixty-two different component samples representing 19 vendors were aged and seismically tested.

The results were favorable. No degradation mechanisms were found that could cause the common-mode failure of these integrated circuits during a seismic event.

A few components were inoperable following either the aging or the seismic testing. A failure analysis was performed that showed the cause of inoperability was unrelated to the aging/seismic stress. The failures, however, do emphasize the need for proper handling of these types of components.

Vendor and component technology remain the significant variables in establishing aging/seismic correlation. As technologies and new vendors emerge and are considered, their aging/seismic sensitivity should be addressed.

References

1. EPRI NP-5024. "Seismic Ruggedness of Aged Electrical Components," January 1987.
2. Jabs, R. H., and Miller, R. B. "Results of a Long Term Aging Program on Electrical Components Used in Nuclear Power Plants," Proceedings of the International Topical Meeting on Operability of Nuclear Power Systems in Normal and Advanced Environments, September 29-October 3, 1986.
3. Intel, Components Quality/Reliability Handbook, 1986.
4. Toshiba, Toshiba Semiconductor Reliability Handbook.

TABLE 1

INTEGRATED CIRCUIT FAILURE MECHANISMS

COMPONENT AND MECHANISMS	CLASS	ACTIVATION ENERGY	FREQUENCY
IC DIE			
Silicon, Silicon Oxide & Interface			
Surface Charge Accumulation	Electrical	1.0 - 1.2 eV	N/A
Slow Trapping, Charge Injection	Electrical	1.3 - 1.4 eV	N/A
Contamination	Chemical	1.4 eV	S
Oxide Defects	Chemical/Electrical	0.3 eV	Y
Dielectric Breakdown	Chemical/Electrical	0.35 eV	Y
Metallization			
Corrosion	Chemical	0.3 - 0.6 eV	Y
Electro-migration	Chemical	0.5 - 1.2 eV	S
Thinning	Mechanical	---	S
IC PACKAGE			
Bonds			
Intermetallic growth (Al/Au)	Chemical	0.3 - 0.6 eV	Y
Bond Fatigue	Mechanical	1.0 eV	S
Wire Fatigue	Mechanical	---	N/A
Lead and Lead Frame			
Contact Degradation	Chemical	1.8 eV	S
Lead Corrosion	Chemical	0.3 - 0.6 eV	S
Fatigue	Mechanical	---	N/A
Die Attach Failure	Mechanical/Chemical	---	N/A
Cavity			
Seal Leaks	Mechanical	---	N/A
Particle Contamination	Mechanical	---	N/A

Y = MAY OCCUR
S = SELDOM FOUND
N/A = NOT APPLICABLE TO AGING AND SEISMIC
--- = NOT AGE RELATED

TABLE 2

COMPONENT FAMILIES TESTED

Component Families	Samples Tested	Vendors Represented	Component Families	Samples Tested	Vendors Represented
Bipolar Schottky	12	4	LSI NMOS	4	2
High Performance CMOS	11	5	NMOS Memory	7	6
Linear Interface (A/D, S/H)	8	5	CMOS Memory	4	4
Hybrid	1	1	EPROM	7	5
Linear	3	3	EEPROM	1	1
NMOS Microprocessor	3	2	EAROM	1	1

TABLE 3

COMPONENTS WHICH WERE INOPERABLE OR ALTERED

ITEM	INTEGRATED CIRCUIT DESCRIPTION	QUANTITY UNACCEPTABLE	ACCUMULATED TEST STRESS	REASON FOR ANALYSIS	OBSERVATION
1	A/D Converter	2	Aging	Exceeded manufacturer's specification on error	No physical anomalies observed One component had poor wire bonds
2	Programmable Peripheral Port	1	Aging	Defective - open circuit	Not dissected - random failure
3	EPROM	1	Aging	Bit pattern change	Not dissected - random failure
4	Schottky Logic	1	Aging & Seismic	Excessive input current	Input transistor shorted from electrical overstress
5	EPROM	1	Aging & Seismic	Excessive input current	Resistive conduction induced by electrostatic discharge
6	EEPROM	1	Aging & Seismic	Change in checksum	40-micron titanium spot on die from manufacturer; no other physical anomalies

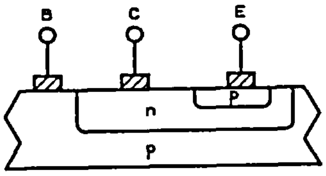


Figure 1
Transistor Cell
Bipolar Technology

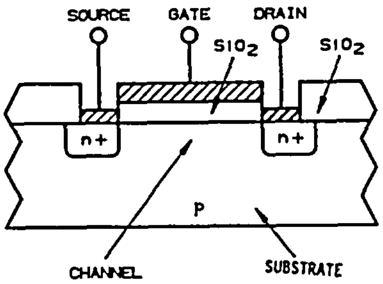


Figure 2
Transistor Cell
NMOS Technology

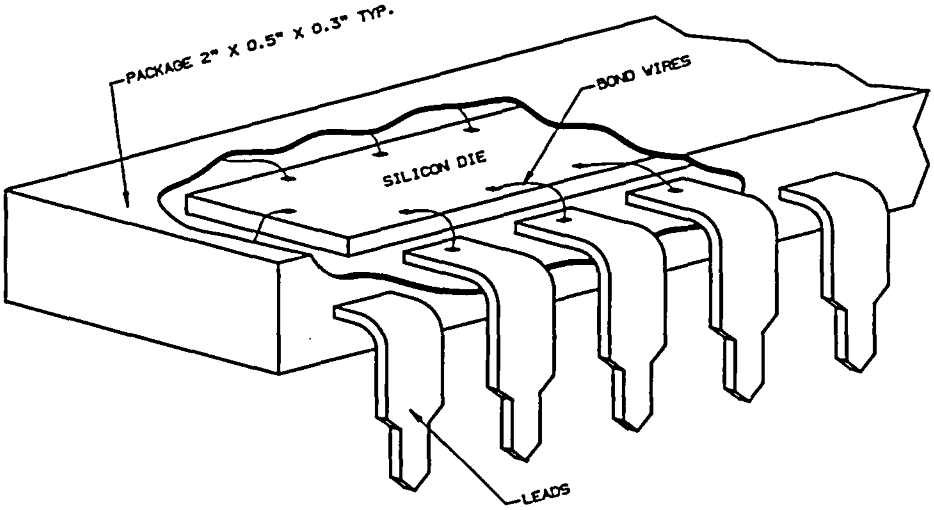


Figure 3
Construction Features of an Integrated Circuit

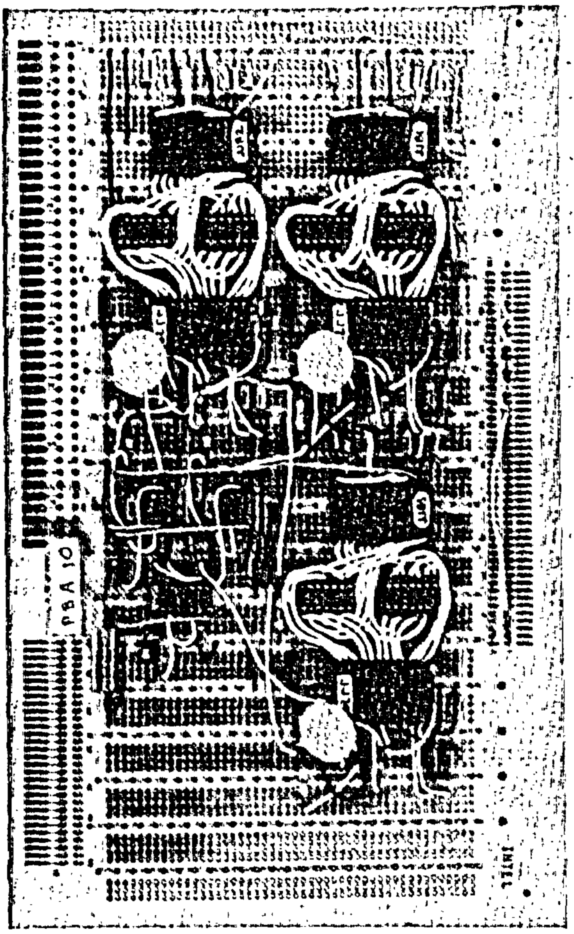


Figure 4
Aged Integrated Circuits
Mounted for Seismic Testing

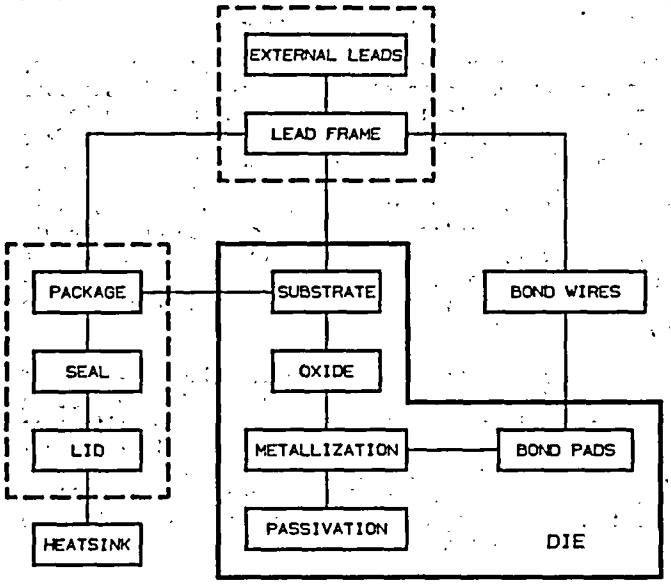


Figure 5
Major Components and Material Interfaces
of an Integrated Circuit

APPENDIX

FAILURE MECHANISMS IN INTEGRATED CIRCUITS

To identify those subcomponents of microprocessor and associated integrated circuits most likely to fail during a seismic event it is necessary to determine the primary material blocks and the interfaces involved. These are shown in Figure 5.

The leads of an integrated circuit handle the electrical and mechanical interface with the rest of the system. The external leads (the part outside the package) are exposed to the operating environment. Failure mechanisms of these leads would be attributed to corrosion, which would not be expected to occur in mild environments of nuclear power plants. If corrosion does occur, it may often be detected by visual inspection.

In integrated circuits packaged in ceramic, the lead enters the package through a glass seal. Mechanical stress on the leads can lead to cracking of the seal and thus a loss of hermeticity. In plastic packages the encapsulating material surrounds the lead frame. Mechanical stress can cause micro-cracks around the leads, allowing moisture to enter. New plastics have been designed to reduce this problem by holding the leads in compression. In any case such damage will allow moisture and corrosives to easily reach the integrated circuit die and cause failure. Temperature cycling followed by a humidity test is used to check for susceptibility in this area.

Inside the package the leads are connected to bond wires that make the connection to the die. Bond failures will be discussed later.

The semiconductor chip is mounted on a ceramic substrate, in hermetic integrated circuits, or a metallic paddle, in plastic parts. Attachment is made with a low-melting-point solder, glass, or epoxy bond. The strength of the bond may be reduced by intermetallic formation or process defects. Defects in the bond lead to local hot spots or an overall temperature rise, and degrade the part's reliability. If the die bond is completely broken because of an external stress or by the pull of the bond wires, failure will occur quickly because of thermal overload. Die attach integrity can be evaluated by destructive die pull tests, radiography, centrifuge, and other common screens.

The die substrate is covered with one or more levels of oxides, metallization and passivation. Defects in the oxide and passivation layers become gathering sites for the ions which lead to electrical failure. The mass of these layers is very small, so they will not be significantly affected by vibration stresses. The metallization, however, may be damaged by mechanical stress, particularly if weakened by another mechanism.

Metallization failures can be precipitated by defects in the aluminum-to-silicon interface. These defects can arise from inadequate control during the alloying process. Overalloying results in a non-uniform current flow, which will lead to an electrical type of failure. Mounds of aluminum called hillocks can form on the metallization because of an excessive mismatch in temperature coefficients of expansion between the metal and oxide. If the hillocks reach a sufficient size, they will break down the oxide or passivation layers leading to electrical failure.

Metallization may fail because of process defects, electromigration, corrosion, or mechanical stresses from adjacent materials. An open circuit may be induced by vibration stress, if the metal or interface has degraded. These failures are more apt to occur at oxide steps where the metal naturally thins to some extent. Step coverage quality is achieved and maintained by an application of good integrated circuit design practice and a well-designed and -controlled fabrication process. Metallization quality is generally evaluated by internal visual inspection and the typical reliability screens.

As is the case with oxide and passivation, the mass of the metallization is small, so the effect of vibration stresses will be limited. However the metallization at the bond pad area provides the mechanical as well as the electrical connection to the bond wire. This means that a significantly greater stress will be placed on this region during mechanical shock because of the larger mass anchored there.

One common connecting scheme uses gold wire attached to the aluminum metallization with a thermocompression bond. However, when gold and aluminum are in contact, an intermetallic, $AuAl_2$ (known as purple plague), forms rapidly at high temperatures. The plague is brittle but strong, so it usually does not break. Failure is generally caused by the voids that form under the bond because of the differing rates of diffusion of gold and aluminum. Experience has shown that the gold/aluminum system is a reliable bond at temperatures below 150°C.

The intermetallic is avoided by using aluminum bond wires and ultrasonic bonding or more complex material systems. More complex systems are used when high-temperature operation is required and a higher price is acceptable. These generally involve various barrier metals and alloys.

Defects that occur during the bonding process often result in degraded reliability. Overbonding or applying too much force or temperature during the bonding process can can damage the bond pad or the underlying silicon, resulting in a weak bond. Underbonding results in less bond strength than desirable. Other defects can scratch the bond pad or nick or draw the bond wire, thus reducing resistance to mechanical and electrical stress. If any of these defects are present, vibration or temperature stress may lead to an open bond. These failures can be accelerated by centrifuge or temperature cycling tests, or a sample from the production lot can be subjected to destructive bond pull to ensure sufficient strength of the bond and wire.

These reliability problems may also occur at the connection of the bond wire to the lead frame. However, intermetallic formation will be less likely since the post temperature will generally be much lower than the temperature of the die. Good bond pads and wires should easily withstand a large amount of vibration stress since bond attach strength is typically tens of thousands of times greater than the mass of the bond wire.

VLSI components are generally packaged in hermetically sealed ceramic or plastic- encapsulated assemblies. Ceramic packages generally consist of two or more ceramic layers connected by glass seals. In some cases a metal lid will be soldered to the package over top of the cavity. The ceramic package has been used successfully for many years with a low incidence of seal failure. When failure of the seal occurs, it is usually precipitated by the stress from the leads where they pass into the package. Various socketing and restraining methods reduce these stresses.

Another failure mode common to all cavity packages is failure due to extraneous particles. Various particles may be left in the integrated circuit package as residue from the die-scribing operation or bonding operation, or that entered from the environment prior to sealing. These particles may not cause any problems until a mechanical stress causes the debris to settle between metal lines or over a gate, where it can lead to a short or an electrical failure. Foreign particles can be detected by internal visual inspection and PIND testing, with some success.

In plastic-encapsulated packages the lead frame, die, and bond wires are all surrounded in epoxy. Therefore, the bond wire will not be free to move, and there will be no problem with loose particles. As previously stated micro-cracks can open along the lead frame, but this is generally due to repeated temperature stressing and not short-term vibration or shock.

Acknowledgment: The authors wish to acknowledge Ron Heller of the Westinghouse Semiconductor Control Center for providing the information for this appendix.

MANAGING DIESEL GENERATOR FAST-START INDUCED AGING

K. R. Hoopingarner and A. B. Johnson, Jr.

Pacific Northwest Laboratory

ABSTRACT

Research on aging of diesel generators by the Pacific Northwest Laboratory (PNL) shows that much of the aging-related wear could be reduced if the engine LOCA emergency starting requirements were extended from 10 to 12 seconds to 30 seconds or longer and the test start and loading times were extended to several minutes. A modest increase in the start time results in several significant advantages, the most important being improved diesel mission reliability and a corresponding public risk reduction. A second advantage is the ability to manage and reduce fast-start induced aging effects.

The PNL diesel generator aging study has developed data and information supporting an integrated diesel generator management program which integrates the elements of testing, inspection, monitoring, trending, and maintenance. The proposed management program would be relatively easy to establish and it is practical for long-term results. These program elements result in improved management of aging and diesel generator operability compared to the current monthly fast-start testing program.

Aging effects on diesel generator availability is the subject of research sponsored by the Nuclear Plant Aging Research (NPAR) Program. The NPAR Program operates under the United States Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

SUMMARY

Diesel generators were selected by the U.S. Nuclear Regulatory Commission to be studied by the Pacific Northwest Laboratory (PNL) under the Nuclear Plant Aging Research (NPAR) Program. The NPAR program is intended to evaluate and recommend ways to identify, monitor and manage aging of nuclear power plant safety-related equipment.

Currently, nuclear diesels are generally tested under fast start/fast load conditions, requiring full speed, full voltage in 10 to 12 seconds and full load in approximately 30 seconds. The aging research program has developed data that confirm that these aggressive conditions cause accelerated aging of the diesel engines. These test conditions were based on the response needed for the loss-of-coolant accident (LOCA) event.

During the performance of the aging research, it was necessary to compare how the observed aging failures interacted with the diesel generator safety mission. The perceived safety function, or mission, has changed since the emergency power systems were originally designed. The NRC has made a major change to 10 CFR 50 General Design Criterion 4, which accepts leak-before-break methodology. Simply put, the need for fast-starts is reduced while the need for long-term secure power after an accident is perhaps increasing. The rationale that PNL developed was that the data and utility experience reviewed did not support the present diesel generator testing methods. The increased wear and failure potential from testing-induced-aging-stressors was not justified by the statistical data obtained. This paper outlines a new management approach that reduces fast-start induced aging and outlines the requirements for a better reliability program for diesel generators.

FAST-START REGULATORY REQUIREMENTS

In 1975, the U.S. regulatory staff were made aware of numerous failure-to-start problems for diesel generator systems. It appeared that the diesel generator reliability could be improved, if assurance-of-starting was addressed. NRC Technical Report, "Diesel Generator Operating Experience at Nuclear Power Plants," OOE-ES-002, June 1975, documented the problem and was part of the basis to develop requirements for Regulatory Guide 1.108, Revision 1, released in August 1977. Revision 1 related testing frequency to the number of failures per 100 tests.

Failure-to-start problems declined as utility and U.S. NRC experience helped to identify failure causes and appropriate solutions were implemented. However, other reliability problems persisted and there was a regulatory awareness of increasing wear and aging problems. In 1982, the Advisory Committee on Reactor Safeguards (ACRS) was advised by the NRC staff that the three-day test interval should be eliminated, and that identifying unreliable diesels and performing necessary repair were more appropriate than promoting additional testing according to the requirements of Regulatory Guide 1.108.

Regulatory Relief of Fast Starts

U.S. NRC Generic Letter 83-41, "Fast Cold Starts of Diesel Generators," December 15, 1983, requested information on detrimental effects of "fast, cold" diesel testing. In 1984, the U.S. NRC issued Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," which requested that licensees take action to change technical specifications to permit slower test starts while retaining the emergency fast-start response requirements. This was followed by a meeting on April 30, 1985, in Bethesda, Maryland, which was intended to obtain information from the vendors and utilities on diesel reliability. Fast-starts and testing requirements were an important part of the discussion during this meeting. Although both the Generic Letter 84-15 and the meeting were informative, they did not change regulatory requirements. In that sense, the aging effects caused by fast-start testing have not been addressed.

The Virginia Electric Power Company (VEPCO) North Anna station requested one of the first technical specification changes for the diesel generator system responding to Generic Letter 84-15. This request was approved by the NRC.[1] In support of their request for a technical specification change, VEPCO and the diesel manufacturer (Fairbanks Morse Engine Division of Colt Industries) identified significant contributors to their engine failures to be: 1) fast starting, 2) fast loading, and 3) test frequency.

Fast, ambient temperature starts are still required for monthly testing of diesels by plant technical specifications and regulatory requirements for the majority of the U.S. Nuclear Stations. Most stations with modified technical specifications do not have the necessary hardware modifications to permit both slow-start testing and fast-start response to valid plant safety demands. Thus, even though the NRC has approved some technical specification relief for a few plants, most of the diesels are subjected to monthly fast starts that intensify aging problems.

The Washington Public Power Supply System Unit No. 2 has qualified a new diesel governor model that will permit both slow and fast-start sequences to be controlled. They have installed this equipment during a 1988 refueling outage. This is one of only a few such qualified governor hardware installations known to the Supply System staff, intended to help manage the effects of fast-start testing of diesel generator systems.

NRC Generic Issue B-56, "Diesel Generator Reliability"

Generic Issue B-56, "Diesel Generator Reliability," is another ongoing NRC staff action area related to testing, fast starts and other diesel issues. Resolution to G.I. B-56 is nearly complete.

NRC Unresolved Safety Issue A-44, "Station Blackout"

Unresolved Safety Issue A-44, "Station Blackout," is also related to diesel reliability. This issue also has been resolved at this time by the release of Regulatory Guide 1.155, "Station Blackout." This guide identifies the need for a new diesel generator reliability program. The aging research results support the development of a reliability program.

AGING IMPLICATIONS OF DIESEL TESTING

Diesel testing may be compared with automobile starting where more wear is experienced in the first 45 seconds of startup, after being parked overnight, than in eight hours of steady-speed highway driving. Because automobiles are started often, many such starting-wear studies have been performed. In contrast to this experience, large diesel engines were developed for purposes that did not involve a large number of starts compared to running hours. Thus, nuclear service diesel starting-wear studies were not performed.

The application of diesel engines in nuclear service differs markedly from typical diesel service in other industrial sectors. For example, marine diesels operate for weeks or months at a time, but the nuclear diesels operate in surveillance testing over short time spans (about one hour of continuous operation). Therefore, the aging implications of nuclear standby service with its large number of starts and rapid loading were not understood.[2]

Current Fast-Start Testing

Figure 1 shows the startup speed profile of a typical nuclear service diesel engine fast-started to typical regulatory guide and technical specification requirements. Several things should be pointed out as significant. Note that the engine first has to develop its rated speed in about 5 to 8 seconds, if steady synchronous speed is to be achieved in the 10 to 12 seconds required by technical specifications for electrical loading to start. This allows time for the natural and unavoidable speed overshoot and the damping shown to occur. The turbocharger also overspeeds by about 15% and overloads during the fast startup transient.

This happens because the exhaust gas energy builds up during the fast speed ramp shown in Figure 1 due to maximum fuel rack settings and the engine speed overshoot. This energy causes excessive turbocharger speed and loading just as the governor system shuts down the fuel rack settings to avoid the overspeed trip point. This also occurs when a full engine load is quickly put on the unit.

During fast starts engine energy buildup quickly heats up the piston and rings. The thicker water-cooled cylinder liners heat up much slower. The result is heavy wear and scuffing caused by the thermal expansion imbalance of the pistons and liners during this fast-start and quick-load period.

Some of the wear during fast starts and loading of the diesels is caused by lack of lubrication during start-up at critical points. Even with diesel engine keep-warm and lubrication oil circulation systems now used, cylinder liners and the upper valve deck do not get adequate lubrication during the first few minutes of operation. Fast-start and loading tests are severe service compared to steady-load power operation.

Modified Start Sequence Advantages

PNL has defined and transmitted to the NRC in report PNL-6287 modified start sequences which show promise to reduce the aging stressors and fast-start effects described. Two alternative sequences to the fast-start profile of Figure 1 are compared in Figure 2. The first added sequence illustrates the improved speed control when the start-up speed ramp is extended to 30 to 45 seconds from the present 10 to 12 second range. The second added sequence in Figure 2 diagrams a more ideal routine test speed profile, which extends the time period to two minutes or more before power loading is started. Both sequences offer improvements in reducing speed overshoot, speed oscillation, and the threat of an engine overspeed trip. Also, the modified test sequence would reduce non-uniform thermal expansion that causes scuffing and would permit lubricants to reach engine components before they are heavily loaded. Management of the diesel generator system to avoid fast-start induced aging should include the consideration of slower engine loading times.

PROPOSED DIESEL GENERATOR MANAGEMENT PROGRAM

PNL is proposing in this paper a new management program for diesel generators, which is intended to reduce aging effects. The research goal was to identify and mitigate the principal aging stressors by practical methods. However, while PNL's program is focusing on aging, any proposed management practices which may mitigate aging effects also invariably improve reliability and influence operational, training and maintenance considerations nominally outside of the NPAR Program scope. Therefore, for this paper the proposed management plan will include, or relate to, these additional considerations. An assumption implied in the proposed program scope is that regulatory changes and plant technical specification changes necessary to implement aging mitigation will be approved by the NRC.

For the operational phase of nuclear power plants, an integrated diesel generator management program is recommended. The proposed program would incorporate monthly testing, inspections, monitoring, trending and maintenance and other program elements into an overall aging mitigation and reliability oriented approach.

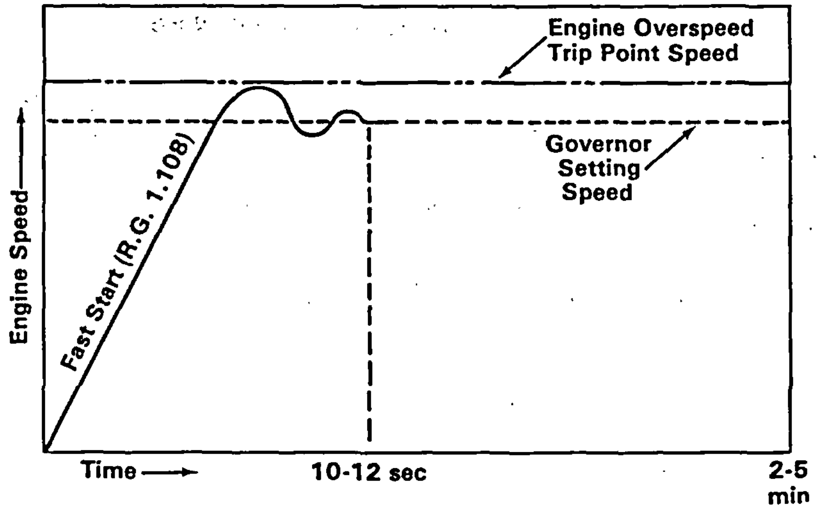


FIGURE 1. "Typical Diesel Engine Speed Profile During Fast-start and Loading Sequences."

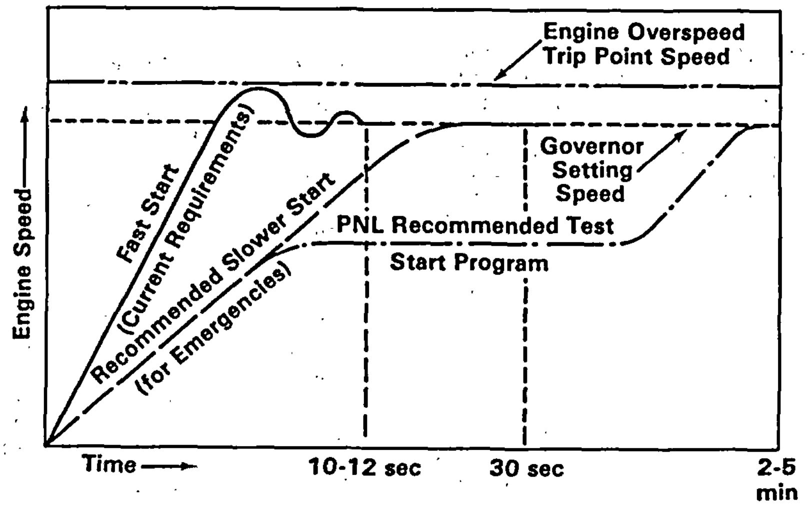


FIGURE 2. "Typical Diesel Engine Speed Profiles Comparing Fast-Start and Modified Start Sequences."

The current statistical basis for monthly fast starts would be changed to a predictive testing program based on the 2 to 5 minute speed profile shown in Figure 2. The purpose of the monthly tests would be to obtain operating data on approximately 50 parameters that indicate current and future engine operability status. These parameters are typically temperatures and pressures, which are measured by currently installed local instrumentation and are similar to the periodic data collected and trended by the small nonnuclear diesel powered utilities operated by some municipalities. A complete listing of these parameters is given in report PNL-6287.[3] Table 1 shows a sample of the parameters of Appendix A of PNL-6287.

PNL recommends that the monthly test period be extended to 4 hours or more. This would allow ample time to achieve stable conditions and collect three data sets of approximately 50 proposed engine and generator operating parameters. Three sets are recommended to ensure no problem indicators are missed. The engines should be operated about one hour before the test data are collected and one hour between each data set. The actual time required to collect the data is about 10 minutes, if two people are used. After the last test period, the engines benefit from a lower-load cooldown period of about one hour. PNL's diesel experts outlined additional benefits for monthly test periods of approximately 10 hours duration.[3]

Monitoring and Trending

The success of the proposed diesel management program depends on the predictive elements of the test approach. Predictability can only be accomplished if the data are collected, analyzed and trended each month as a key program element. Monitoring and trending have two important functions. First, the procedures must be developed to give the diesel operators immediate safety level information. Second, the long-term engine aging effects must be trended.

The first function may be achieved by setting up the data collection sheets to show the normal and safe operating range for each parameter after stable operation is obtained at specified test load conditions.

TABLE 1. Exhaust System Parameters to be Monitored and Trended (Sample of Parameters Recommended)

Parameter	Recommended Requirements Required	Recommended Requirements Optional
Exhaust System[a]		
Exhaust temperature out of each cylinder, Cylinder No. 1 Cylinder No. 2, Etc.	X	
Exhaust temperature to turbocharger turbine (preturbine temperature), more than one thermocouple may be required	X	
Exhaust pressure to turbine[b]		X
Exhaust temperature from turbine	X	
Exhaust back pressure in exhaust pipe after turbine exhaust expansion section[b]		X

(a) Note it is important that the same electrical load be used for each test (e.g., 100% of plant load).
(b) Where equipment exists.

Normal, last data point, alert and engine test shutdown parameters would show on the data sheets used by the operators. At the end of the test, assuming all parameters are in the normal range, near-term diesel-generator operability may be predicted. Deviations from normal indicate that maintenance should be scheduled. Trending for long-term changes and deterioration, in a similar manner, measures aging, wear and other slow-change parameters of the engine and generator system. After trending analysis, longer term operability may be predicted or any necessary repair, replacement, or maintenance may be identified and scheduled.

Engine Disassembly

PNL'S research also investigated the role of maintenance and preventative maintenance on aging mitigation. Some of the results were not expected. For example, engine and component disassembly for inspection purposes did not lower failure rates. One study showed that this type of maintenance on various components did not lower failure rates 80% of the time.[4] Another U.S. Navy study showed that periodic overhaul was not beneficial 73% of the time for various types of components.[4]

For regulatory purposes and safety assurance, prescribed periodic diesel generator disassembly, for inspection, is not supported by the available data. Improved assurance of engine reliability is obtained if maintenance is directed to: 1) repair defects and obvious failures, 2) perform recommended engine service and inspections without engine disassembly, and 3) perform maintenance to correct components identified by the monitoring and trending program. Management of maintenance and maintenance training are viewed as part of overall diesel generator management.

Recommended Program Advantages

The current regulatory requirements were based upon periodic testing of the diesel generators to generate statistical data to indicate reliability. Periodic disassembly of the engines for inspection is also required to further support the reliability goal. PNL's proposed management program is based upon aging studies which show that significant aging effects actually are created by these requirements, and these would be eliminated if reliability is determined by the monitoring and trending methods outlined in this paper. The data from the NPAR program also do not support routine engine disassembly for periodic inspections. Table 2 shows a brief summary of some of the important program elements in the existing and proposed management approaches.

DISCUSSION OF THE RATIONALE AND DATA

The reduction of the plant safety risk due to station blackout, the loss of all AC power, appears to be the proper goal. Several minutes before AC power is resumed poses little plant risk, assuming no LOCA event. Longer station blackouts have increasing risks with increasing time. This supports the rationale that demonstration of future engine operability, the goal of the recommended test program, is the correct approach.

Station blackout risk can be reduced further by a relatively minor modification to allow the operators to manually start the diesels. Report PNL-6287 describes how the operators can be trained to go directly to the engine location and successfully start and load the engines within a few minutes. The hardware changes to permit direct manual starting are simple and do not involve major expense. This manually bypasses several of the highest aging-failure-rate diesel systems and components which may prevent successful automatic startup and subsequent power production.

Statistical analysis of the information obtained by the present test methods also supports the recommended testing approach. Aging assumes that performance deteriorates with time. Given that an engine system has excellent monthly test results and then for a variety of reasons quickly deteriorates (ages) to a reliability that is not acceptable to either the owner or the NRC, it can then be shown, using statistical probabilities, that the overall poor engine condition may not be detected for periods approaching a year. In contrast, the recommended testing approach would more likely detect most of the important incipient problems during the first test period.

TABLE 2. Comparison of Current and Proposed Management Programs

Program Element	Current Program	Proposed Program
Periodic engine overhauls are required for inspection.	Yes	No
Testing program is intended to identify incipient failures.	No	Yes
Testing program is chiefly designed to predict future operability.	No	Yes
Fast-start induced aging and wear are minimized.	No	Yes
Current diesel generator reliability and station blackout issues were a program consideration.	No	Yes
Aging of fuel and lube oil were considered in the program development.	No	Yes
Statistical variations and problems are likely to affect the management program.	Yes	No

CONCLUSIONS

In summary, the key points for advocating the proposed diesel generator management system are:

- The proposed approach offers elimination of some known aging stressors and timely detection of many of the other stressors.
- The proposed approach appears to offer improved diesel generator management benefits for both the utilities and the NRC which should result in improved reliability and increased detection of potential engine failures and degraded conditions.
- The proposed approach can provide many benefits without any regulatory or standards changes. Appropriate regulatory and standards changes could achieve further benefits by avoiding frequent fast starts with their corresponding aging stessors.
- The proposed approach requires few physical plant modifications and within a few years would be lower in cost than the present periodic testing and mandatory engine overhauls.

REFERENCES

1. NRC (Nuclear Regulatory Commission). 1985. Safety Evaluation by the Office of Nuclear Reactor Regulation, Related to Amendment No. 48 to Facility Operating License No. NPF-7, Virginia Electric and Power Company, Old Dominion Electric Cooperative, North Anna Power Station, Unit No. 2, Docket No. 50-339.

2. Hoopingarner, K. R., J. W. Vause, D. A. Dingee, And J. F. Nesbitt. 1987. Aging of Nuclear Station Diesel Generators: Evaluation of Operating and Expert Experience, Phase I Study. NUREG/CR-4590, Vol. 1. Prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.

3. Hoopingarner, K. R., B. J. Kirkwood, and P. J. Louzecky. 1988. Study Group Review of Nuclear Service Diesel Generator Testing and Aging Mitigation. PNL-6287. Prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.

4. Prichard, J. W. 1984. Equipment Age-Reliability Analysis. Naval Engineers Journal, pp. 65-70.

5. U.S. NRC. 1983. Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability. Generic Letter 84-15, Nuclear Regulatory Commission, Washington D.C.

EPRI CABLE AGING RESEARCH

G. E. Sliter
Electric Power Research Institute

Summary. This paper describes two ongoing cable aging research projects sponsored by the Electric Power Research Institute and the role played by these projects in the development of technology that may prove useful for justifying operation of nuclear plant cable systems beyond their current qualified life of 40 years. The first project is generating material property data on cable specimens placed in nine operating reactors. The material property data and corresponding environmental measurements of specimens withdrawn over coming decades will be compared with measured degradation in artificially aged specimens to quantify the conservatisms in traditional qualification practice. The second project is developing a hand-held, nondestructive indenter to assess cable aging by means of in-situ measurement of compressive properties of polymeric jackets or insulations. Comparison of measurements made on installed cables with previously developed baseline indentation data can determine the state of aging of installed cables. Both projects will be valuable in nuclear plant life extension programs.

Introduction

Safety-related power, control, and instrumentation cables in nuclear plants have been manufactured and qualified by industry standards[1,2] taking into account the degrading effects of thermal and radiation aging. The traditional practice has been to subject cable samples to accelerated aging tests and then monitor their operability during and after the application of simulated accident conditions. Almost all safety-related cable has been qualified for the current 40-year licensed term of plants. It is not surprising, therefore, that cables in plants, which have been in operation 28 years or less, have been very reliable. An NRC-sponsored survey of operating experience[3] identified only a handful of failures attributable to aging.

On the other hand, both accelerated thermal aging based on Arrhenius theory and the equal-dose/equal-damage assumption of accelerated radiation aging have many limitations.[4] NRC-sponsored research[5] has examined the uncertainties due to synergisms and dose-rate effects on the life predicted by traditional qualification practice. This practice has made up for such uncertainties by using conservative estimates of operating and accident environments and by applying proof test margins specified in IEEE standards. As plants grow older, questions arise concerning the continued ability of cable systems to perform adequately. What effects do the uncertainties in the accelerated aging approach have on actual useful life? Are there any unanticipated aging stressors and age-related failure modes? Are there localized areas in plants where the ambient environments are more severe than assumed in qualification? For example, are there hot areas near pipes or deep within cable bundles perhaps surrounded with fire protection materials? Can we correlate age degradation with a cable's ability to perform under accident conditions? Reference 6 is a good summary of cable life issues.

Pilot nuclear plant life extension studies sponsored by EPRI[7,8] identified cable systems as "critical" components in that a utility's ability to address the issue of evaluating the effects of significant age-related degradation of cable systems is crucial to plant life extension. It would be costly--at least tens of millions of dollars--to replace the many miles of safety-related cables in a single nuclear unit.

The Electric Power Research Institute (EPRI) recognized that uncertainties about whether current procedures adequately account for all significant in-plant degradation due to aging can lead to excessive costs for plant surveillance, to exaggerated estimates of safety risks, and to pessimistic estimates of cable or plant life. This motivated initiation of a cable aging research program several years ago. This paper describes the objective, scope, and status of two major projects in the EPRI nuclear power cable research program: an in-plant study for comparing natural versus artificial aging and the development of a hand-held indenter for monitoring the aged condition of installed cable. It also indicates other current and planned cable research by EPRI and how it relates to research by others.

In-Plant Aging Study

To assess artificial aging theory and practice and to allow improved lifetime predictions for cable and equipment, EPRI is sponsoring an in-plant and laboratory aging study by the University of Connecticut's Institute for Materials Science. Cofunding is being provided by Northeast Utilities and Detroit Edison.

There are other programs with similar objectives. Two are being conducted by U.S. utilities.[9,10] Another applies to nuclear plants in Germany.[11]

Approach

The EPRI program consists of placing specimens of widely used types of cable and electrical components in the reactor buildings of operating nuclear plants for from ten to forty years or longer. Although the specimens include small electrical components, parts such as O-rings, and lubricants, this description focuses on the cable specimens. The specimens are not energized due to limitations in available resources. One of the major effects of being energized is simply an increase in temperature due to self heating. This effect can be addressed by interpreting the ambient temperatures at some of the hotter locations in the participating plants as being equivalent to a cooler ambient plus a temperature rise due to self heating.

Utility personnel will remove the specimen bundles and environmental monitors during planned outages. Researchers will measure physical properties of the materials and compare the degradation of the naturally aged materials with that of identical specimens aged artificially under equipment conditions as they would be in a qualification test program. It is presumed that quantifying differences in material property degradation between artificial aging and natural aging is a valid way to assess any differences in functional performance that may occur due to differences in aging. It is the condition of a component that is simulated by accelerated aging methods used in nuclear plant qualification practice. Subsequent to accelerated aging, the component's functional performance is checked during and after simulated adverse environments.

To reach generic conclusions it was necessary to include a spectrum of cable types, plant types, as well as specimen locations in the participating plants.

Description of Cable Specimens

The types of cable in the program consist of "generic" types and "plant-specific" types. Generic specimens (see Table 1) are the four types placed in all locations in the original eight participating plants. These types are widely used in the nuclear industry as determined from the EPRI Equipment Qualification Data Bank[13] and represent a broad cross-section of manufacturers, materials, applications and constructions. Each participating utility submitted one cable type for placement in its plant only. These plant-specific cables broadened the scope of specimens to cover five additional manufacturers and several additional cable constructions.

Recently, cable types in this program have been expanded by adding three of the twelve types being aged and accident tested in an NRC-sponsored program at Sandia National Laboratories.[14] The types added are:

- Rockbestos, 22 AGW coaxial, XLPE insulation, shielded
- Rockbestos, 1C 16 AWG, 600V, silicone rubber insulation, fabric jacket
- Champlain, 1C 12 AWG, polyimide (Kapton^{tm}) insulation, unjacketed

Placement of these at five existing locations in three participating plants will assure that several cable types are common to the EPRI and NRC programs.

All cable specimens are 1-foot lengths with shrinkfit end caps to prevent gas and moisture intrusion.

Plants and Environments

Fifteen specimen "bundles" (see Figure 1) were placed at each of fifteen locations in eight plants during 1985. As indicated in Table 2, five of the plants are PWR's and three are BWR's with inerted atmospheres. All of the bundles are in reactor containment areas except one located in the steam tunnel of a BWR.

The specimen locations were selected to give a reasonably wide range of environments. Ambient temperatures and radiation levels measured to date are shown in Table 2. Average temperatures range from 78°F to 131°F. Forty-year doses range up to 3 Mrads.

For the most part, these measured levels are substantially less than levels estimated by the utilities on the basis of the bounding levels used in qualifying equipment. To obtain data at more severe aging conditions, arrangements have been made to place five additional specimen bundles at each of two "hotter" locations in one of the containments at Virginia Power's Surry plant. Temperature at these locations is estimated to be as high as 180°F and total 40-year dose is at least 120 Mrads. These specimens are expected to lead all of the others in providing useful degradation data.

Figure 1. Wire Mesh Specimen Bundles Hanging at One of Two Locations in Commonwealth Edison's La Salle Unit

Property Tests

Physical property tests include measurements of mechanical properties (tensile elongation-to-break, strength and modulus) and density. The tensile tests are performed on thin strips (0.25 mm or less thick and 2.5 cm long) which are cut from the cable specimens with a special microtome instrument.

To date bundles have been returned from the plants and tensile/density measurements have been completed on about half of the returned cables. With maximum aging durations of only three years there is still no degradation in properties within the accuracy band of the measurements. It is expected that natural age degradation appreciable enough to begin being useful for constructing models and comparing with artificial age degradation will not occur until about 1995, ten years after initial placement.

Mathematical Aging Model

The mechanical property data on naturally aged specimens will be used to construct an empirical mathematical model of aging; i.e., a formula for damage (property degradation) as a function of time and environmental conditions. The model will be constructed by using statistical analysis to fit a postulated functional form with the measured properties of specimens from many plant locations. This model will be useful for gaining an understanding of the processes contributing to age degradation. It will also go beyond the Arrhenius theory in the model's ability to predict degradation as a function of time and to account for the combined effects of temperature, radiation and humidity.

Table 1

Generic Cable Types in Specimen Bundles

Manufacturer	Type	Insulation	Jacket	No. of Conductors	Size of Conductors
BIW	Instrument	EPR[a]	CSPE[b]	2	16 AWG
Kerite	Power	EPR	CSPE	1	6 AWG
Okonite	Control/Power	EPRI/CSPE	CSPE	3	10 AWG
Rockbestos	Control	XLPE[c]	Neoprene	3	12 AWG

[a] Ethylene propylene

[b] Chlorosulfonated polyethylene (Hypalon^tm)

[c] Cross-linked polyethene

Table 2

Measured Environments at Specimen Locations in Plants (NA = Not Yet Available)

Plant (Utility)	Type	Average Ambient Temperature (°F)	40-Year Dose (Mrad)
DC Cook 1 (American Electric)	PWR	78 113	NA 0.4
Maine Yankee (Maine Yankee Atomic)	PWR	104 106	0.8 NA
Millstone 2 (Northeast Utilities)	PWR	130	1.4
Point Beach 2 (Wisconsin Electric)	PWR	100 86	NA 0.1
Trojan (Portland General Electric)	PWR	131 130	<0.1 3.0
LaSalle 2 (Commonwealth Edison)	BWR	113 107	0.6 0.2
Peach Bottom 3 (Philadelphia Electric)	BWR	NA NA	NA NA
WNP-2 (WPPSS)	BWR	115 117	2.5 <0.1

Artificial Aging

The same property tests conducted on specimens removed from plants will be performed on artificially aged specimens. The aging parameters required for comparison to the results of natural aging in this study have been developed for each specimen type in consultation with its manufacturer and with Franklin Research Center. Standard aging practice (thermal and radiation) is being used. Accelerated aging parameters have been selected to be representative of both the actual (measured) in-plant conditions and the conditions that would have been used for qualifying equipment in the plant locations (i.e., design temperatures and doses). Cable samples have been aged at a temperature of 132°C (269°F) for four durations up to 47 days. Based on an activation energy of 1.135 eV, this simulates 40 years of thermal aging at an average containment temperature of 71°C (160°F). Radiation aging will be completed by the end of 1988.

Comparison of Natural and Artificial Aging

Because it is not possible to match each naturally aged data point (from a specific specimen, location, and exposure duration) with a corresponding artificially aged data point, a method has been developed to compare natural/artificial degradation data and interpret any differences noted over a broad range of parameter space (average temperature, dose, dose-rate, and time).

First, as mentioned above, the specimens receive accelerated thermal aging at a single temperature normally used for qualification aging. Specimens are extracted periodically from the aging oven and their physical properties measured.

Next, specimens thermally aged to various levels receive accelerated radiation doses that span the expected range of plant doses for a 40-year-period (doses of 5 and 20 Mrads are planned).

Degradation histories of artificially aged specimens are then constructed as graphs of measured property versus accelerated time for each value of dose. (Scatter in the measurements will require some type of statistical fit of the data.) With the Arrhenius equation and the equal-dose/equal-damage assumption, these graphs (with interpolation for dose) will predict a material property degradation for any value of dose within the range applied in the laboratory and for any plant service temperature. In this manner, values of predicted and measured degradation can be compared for all plant specimens.

Results will be used to evaluate artificial aging theory, quantifying to the degree possible the conservatisms that Standards writers intended to build into qualification practice. This research could also generate plant-specific data to justify continued service of cabling beyond its current 40-year life; or it could forewarn plant operators of greater-than-expected degradation if any is observed.

Cable Indenter Aging Monitor

Greater-than-expected cable degradation could also be detected by means of an in-situ, nondestructive mechanical condition test of insulation and jacket material. Another EPRI cable program is developing a hand-held cable indenter device for performing such tests. The methodology can serve as a field check of actual cable condition compared with the condition predicted by qualification aging programs. It can also provide in-situ data to support justification for extending the qualified life of cable systems beyond 40 years.

Traditional electrical tests, such as insulation resistance and high potential testing, are not sensitive enough to detect the level of age-related deterioration at which a cable can no longer withstand the effects of an accident environment. Present electrical tests suitable for in-service unshielded cable may not even detect aging-induced cracks that penetrate to the conductor if the cable is dry.[3] Therefore, measurement of the mechanical properties of cable polymers is the best way to track the vulnerability of cables to age-induced cracking, which would lead to electrical failure in a moist accident condition. The cable manufacturing industry has found elongation-at-break to be a useful test for evaluating cable aging both from service or from laboratory age-conditioning. Unfortunately, testing for elongation-at-break is destructive and must be performed in a laboratory. Removing cable samples from service to perform such tests is not desirable. A nondestructive test that can be used in-situ is needed.

EPRI is sponsoring the development of a cable indenter aging monitor by the Franklin Research Center.[15] The method being developed uses an instrumented anvil that is pushed against the surface of the cable jacket or insulation (Figure 2). Therefore, the depth of penetration for a given force will decrease as the cable materials age. The indenter is a quantified version of a troubleshooter's practice of testing the hardness of a cable by pressing a fingernail into it. If the indenter test is performed periodically, the results will provide an indication of used and remaining cable life.

To permit application of the test method, benchmark data may have to be developed for each cable material and geometry. However, the cost of obtaining these data appears small when compared to the cost of the present alternative--removing cable from service for destructive testing.

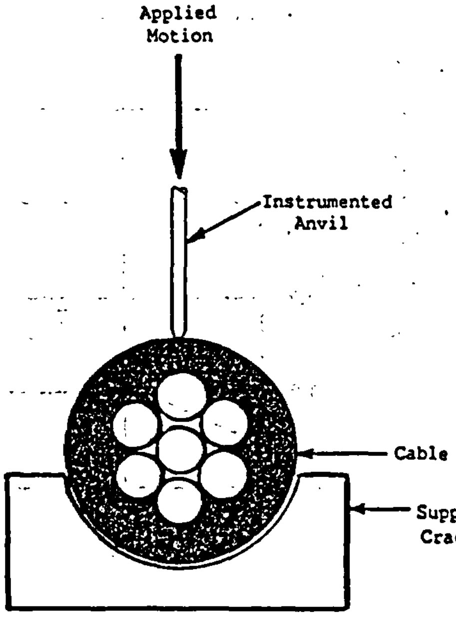


Figure 2. Diagram of the Concept for a Nondestructive Cable Insulation Test Device

The work performed to date has demonstrated the technical feasibility of the indenter methodology. The following subsections describe the feasibility study and the follow-on efforts to bring the technique to a usable stage.

Feasibility Study

Researchers used a laboratory compression rig (Instron test machine) to record the indent behavior of several cable types that had been thermally aged to levels less than, equivalent to, and greater than the age conditioning used in many qualification programs. The parameters that were evaluated as potential indicators of aging effects were:

- resistive force during application of fixed-rate deformation
- depth of penetration for a predetermined applied force
- relaxation of resistance force with fixed applied deformation
- creep or continuing deformation with fixed applied force.

The first of the parameters, the force characteristic during fixed-rate deformation, has shown the greatest potential. The rate of increase of force during deformation (the slope of the force-time curve) correlated well with cable aging (Figure 3). This parameter is equivalent to the force-deformation slope, or modulus. Indent tests performed on several types of artificially aged materials showed a similar significant increase in the modulus with increasing age. The same trend was observed for ethylene propylene rubber (EPR) insulation and chlorosulfonated polyethylene (CSPE) jacketed cables of different types and sizes. The trend was neither regular nor repeatable for cross-linked polyethylene (XLPE) insulation.

Cables tested included large single-conductor power cable, and single- and three-conductor control cables, all of EPR-CSPE construction. The XLPE insulation alone (cable stripped of CSPE covering) showed no usable mechanical property that tracked with aging. However, the CSPE jacket, which does change mechanically with age, can be used to track the cable's aging.

Tests were also performed for three ages of the three-conductor control cable. Readings were taken at six points around the circumference of the cable of each age to determine if the core configuration caused a significant variation in the results of the test. While variations were observed, indicating that the position of the filler and conductors under the jacket did affect the results, the differences were not gross. Reasonable confidence exists that the test method can be usefully applied to multiconductor cable jackets.

Comparisons were made between indenter test data and standard durometer hardness measurements. The response from indenter tests showed a markedly superior sensitivity to aging and ease of application to plant cables.

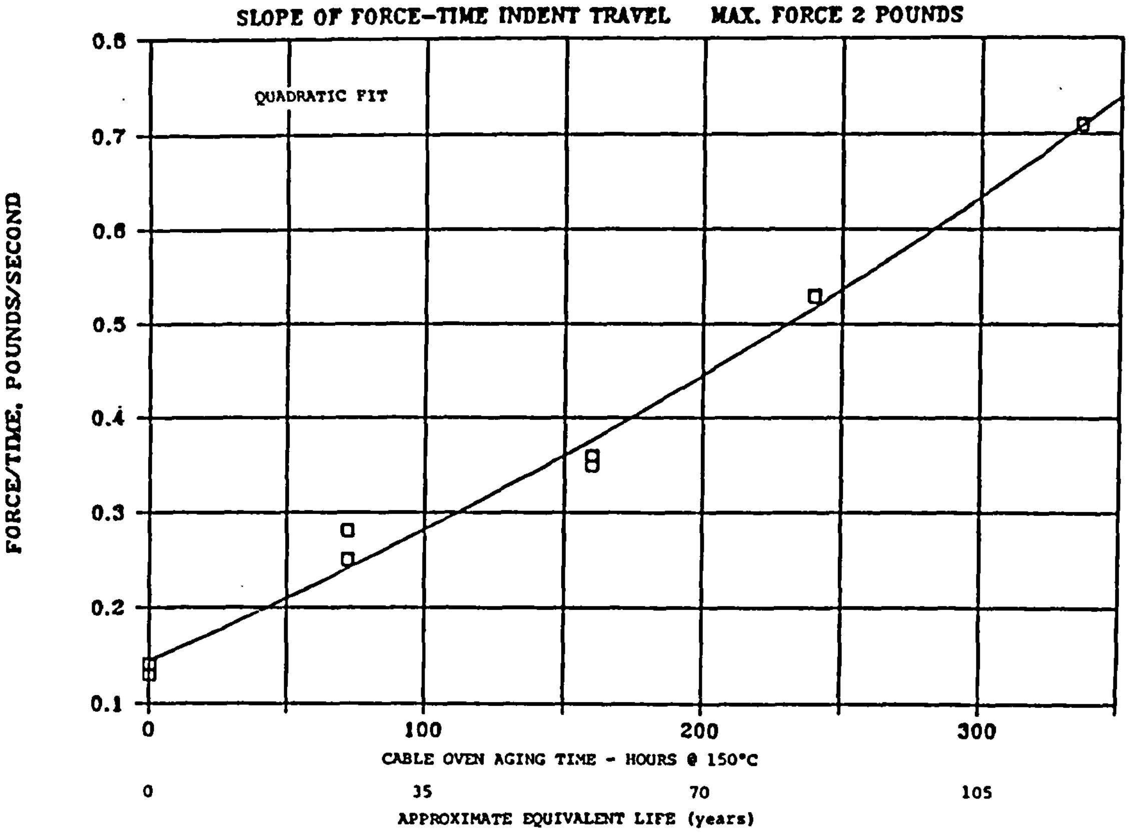


Figure 3. Slope of Compression History for 500-MCM EPR/CSPE Cable During Constant Rate Penetration

Development of Criteria for Evaluating Plant Cable

To determine if cable is acceptable for continued service, criteria that take into account continued normal service capability coupled with accident service capability are necessary for evaluation of the test results. As part of the environmental qualification of the cable for nuclear power plants, each type of qualified cable is normally subjected to accelerated thermal and radiation aging prior to being subjected to accelerated accident conditions. To develop the benchmark condition parameters for in-plant usage, unaged cable samples will be subjected to the accelerated thermal and radiation aging of the qualification program. The results of indenter tests on these aged samples will be used as a benchmark (see the "baseline" curve in Figure 4). Cable in the plants can then be tested by the same means to construct the dashed "in-situ" curve in Figure 4. Using the same material property value that formed the basis for a 40-year qualified life, the in-situ data sketched in Figure 4 would indicate a greater-than-40-year predicted life.

Limitation on Applicability in a Plant

The test method will indicate the condition of the cable in specific locales and therefore, even without the baseline measurements described above, can be a surey tool for indicating the relative condition at various locales. Care must be taken to evaluate the condition of the cable at locales that are subject to different thermal, radiation, or chemical conditions.

Some limitations or difficulties in field application are expected: some cables may be inaccessible, some are located in conduits, and some are in densely packed trays. Some plants have used fire-retardant coatings on cables. The test method will have to be applied where these cables are accessible. Tests results are expected to be useful even when much of the cable is enclosed in conduit because the conduit should help protect the cable and shield it from radiant heat sources. Even though the test is expected to be easy to perform and of short duration, it will probably be used on a sampling basis rather than on all cables because of the large number of cables in a plant. The most logical points for testing would be those where the worst-case normal environments are known or suspected to be.

In combination with baseline data on samples that have been accident tested, the indenter will indicate the cable's ability to withstand an accident condition. However, it will not detect an isolated flaw such as deformation at a pressure point or a corroded shield. Nor will it be useful in detecting such conditions as water treeing that can affect cables with voltages above 5 kV. Such cable defects could be detected by random failures during normal operation or else require other insitu test methods (see next section) for detection.

Follow-on Development

The work described above demonstrated the technical feasibility of the indenting technique. A follow-on program is continuing the development of the technique leading to an engineering model of a hand-held test device by early 1989. Tests are being performed on thermally aged and laboratory irradiated cables obtained from the University of Connecticut program. Additional tests are being performed for different insulation and jacket materials, and on various types of multi-conductor cables. The effect of cable temperature on indenter measurements also needs to be examined. Once the hand-held test device is available,

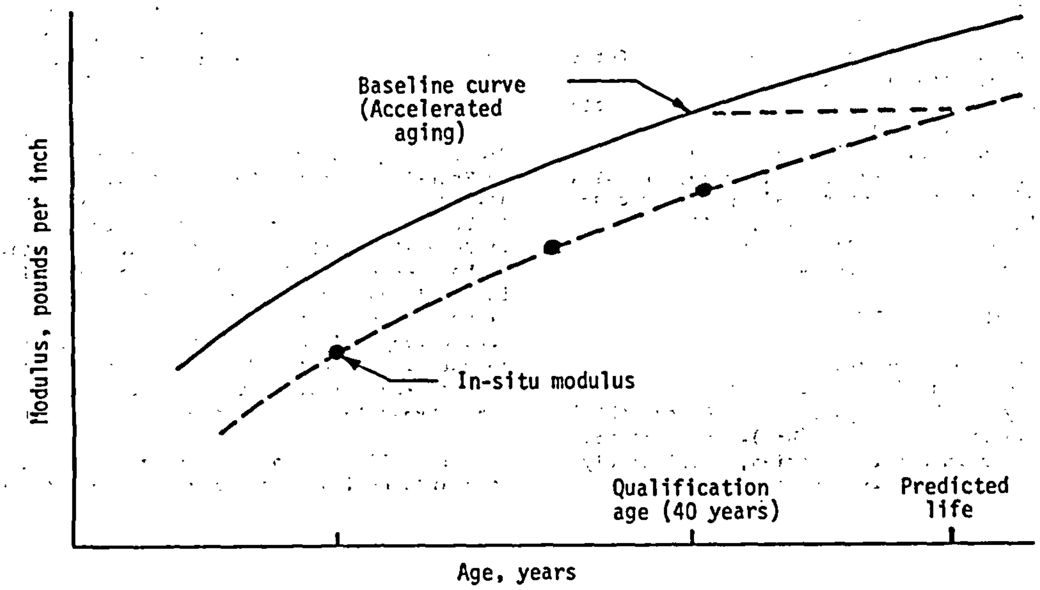


Figure 4. Sketch of Baseline and In-situ Modulus versus Aging Showing How Measurements may be Used to Extend Cable Qualified Life

it will be applied in a nuclear plant to demonstrate the methodology.

OTHER EPRI CABLE EFFORTS

Cable Condition Monitoring

An EPRI Workshop on Cable Condition Monitoring held in February 1988[18] convened experts worldwide to identify the state of the art of cable condition monitoring and diagnostic techniques and to assess the need for research on improved or innovative techniques. One of the important conclusions of the workshop was that it is unlikely that one all-encompassing test method or procedure can both measure the quality of the "hay" (material condition of aged insulation) and find the "needle" (local defects or degradation in insulation). The indenter method described above examines the hay but not the needle. EPRI is initiating a project at Sandia National Laboratories (SNL) to develop practical, in-situ condition monitoring techniques that can be used to detect local degradation produced either by localized wear/aging or by damage during installation or maintenance. The project will attempt to correlate monitored parameters with the ability of the defective cable to perform its safety function during postulated accidents.

One of the leading candidate condition monitoring techniques to be examined in the Sandia program on local cable degradation is an innovative method which creates an ionized gas sheath around an unshielded cable. The ionized gas provides a path for DC breakdown from a defective area to a nearby ground such as a conduit or cable tray. The DC voltage is substantially lower and therefore less damaging than that needed to produce breakdown in a conventional DC high-potential test in which the air surrounding the defect insulates it from ground. Demonstration testing of this technique has been performed and engineering development is now beginning.

Cable Life Extension Guidelines

EPRI plans to sponsor an effort in the near future to provide utilities with guidelines for plant activities that can begin now which will maximize the option for extending the qualified life of cable systems beyond 40 years. The activities include environment measurement and control, baseline monitoring, reevaluation of qualified life based on actual operating conditions, cable inventory and archiving, failure evaluation and trending, degradation mitigation, restorative measures, and recordkeeping.

OTHER CABLE RESEARCH

The EPRI cable research described above complements and is coordinated with other cable research being sponsored by the U.S. Department of Energy (DOE) and Nuclear Regulatory Commission (NRC). Substantial cable aging research in other countries, particularly Japan, Germany, and France, is not covered here.

DOE-Sponsored Research

The EPRI research is being coordinated with that of the DOE in support of the industry Nuclear Plant Life Extension (NUPLEX) program. DOE-sponsored projects include:

- Development of Cable Life Extension Technical Criteria (SNL)

  Addressing the technical bases most appropriate for justifying extended life such as laboratory based qualification testing, cable reliability data bases, condition monitoring techniques, natural aging data bases, redefinition of environmental conditions and replacement of selected cables.

- Environmental Characterization for Critical Cable Applications (co-sponsored by Virginia Power Corporation)

  Performing environmental monitoring (thermal and radiation) at critical cable locations inside the Surry containment that might impact safety and operability during life extension.

- Combined Environments Life Prediction Methodology (SNL)

  Determining the extent to which a "combined environment" life prediction methodology is useful for life extension; the methodology uses research test data to account for dose rate and synergistic effects in determining cable degradation and remaining life for several commonly used insulation materials.

NRC-Sponsored Research

The NRC has sponsored and continues to sponsor extensive research on cable aging, monitoring, and accident performance. Most of the work was performed since the mid-1970's under the equipment qualification program at SNL. Reference 5 is a good summary of the scope, results and conclusions of the program.

Other NRC cable research falls under its Nuclear Plant Aging Research (NPAR) Program. An early effort by Franklin Research Center[3] reviewed cable aging mechanisms, operating experience, and existing monitoring techniques. Another NPAR effort begun recently at SNL is determining the degree to which condition monitoring techniques can predict the performance of aged cables in simulated accident conditions.[14] One of the techniques to be assessed is the EPRI cable indenter aging methodology. An assessment of the life extension potential of about a dozen cable types will be performed, several of which as mentioned above, match cable types in the EPRI in-plant aging study.

The NRC is also sponsoring a National Bureau of Standards review of candidate methods for detecting incipient defects due to aging of installed cables in nuclear power plants.[17] The methods identified as having the most potential for success are being considered for inclusion in both the NRC and EPRI cable condition monitoring programs at SNL.

SUMMARY

The major issues being addressed by cable aging research by EPRI and others are:

1. Differences between artificial (accelerated) versus natural (in-plant) aging
   - assumed versus actual environment
   - validity of Arrhenius equation and assumption of equal dose/equal damage
   - significance of dose-rate/synergistic effects and sequential versus simultaneous application of accelerated stressors
2. The correlation of material degradation and electrical monitoring with performance under operating and accident conditions.

Completion of the current extensive research efforts addressing these and other issues will provide much understanding of how cables age and can guide the implementation of sound, reasonable and cost-effective aging management programs for cable life extension. It is important also that we improve understanding of aging mechanisms in cable connectors and penetration assemblies.

REFERENCES

1. IEEE Standard 323-74 and -83, IEEE Standard for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations, Institute of Electrical and Electronics Engineers, 1974 and 1983.
2. IEEE Standard 383-1974, IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connectors for Nuclear Power Generating Stations, Institute of Electrical and Electronic Engineers, 1974.
3. S. Ahmed, S. P. Carfagno, and G. J. Toman, "Inspection, Surveillance, and Monitoring of Electrical Equipment Inside Containment of Nuclear Power Plants - With Applications to Electrical Cables," NUREG/CR-4257, ORNL/sub/83-28915/2, Oak Ridge National Laboratory, Oak Ridge, Tennessee, August 1985.
4. S.P. Carfagno and R.J. Gibson, "A Review of Equipment Aging Theory and Technology," Final Report EPRI-NP-1558, Electric Power Research Institute, September 1980.
5. L.L. Bonzon, F.J. Wyant, L.D. Bustard, and K.T. Gillen, "Status Report on Equipment Qualification Issues Research and Resolution," NUREG-CR-4301, SAND85-1309, November 1986.
6. L.D. Bustard, "Definition of Data Base, Code and Technologies for Cable Life Extension, Sandia National Laboratories Report SAND86-1897, March 1987.
7. "PWR Pilot Plant Life Extension Study at Surry Unit 1: Phase 1," Electric Power Research Institute, Final Report EPRI NP-5289P, July 1987.
8. "BWR Pilot Plant Life Extension Study at Monticello Plant: Phase 1," Electric Power Research Institute, EPRI NP-5181M, May 1987.
9. T.J. Al-Hussaini, "Cable Condition Monitoring in a Pressurized Water Reactor Environment," "Proceedings of the Workshop on Power Plant Cable Condition Monitoring," Electric Power Research Institute Report EL/NP/CS-5914, July 1988.
10. S. Kasturi and S. Litchfield, "Cable Condition Monitoring Program at Perry Nuclear Power Plant," "Proceedings of the Workshop on Power Plant Cable Condition Monitoring," Electric Power Research Institute Report EL/NP/CS-5914, July 1988.
11. W. Morel and H. Rose, "Radiation Aging of Cables Under NPP Conditions," Proceedings of the International Conference on Nuclear Power Plant Aging, Availability Factor, and Reliability Analysis, San Diego, California July 8-12, 1985.
12. M.T. Shaw, "Natural Versus Artificial Aging of Nuclear Power Plant Components," Electric Power Research Institute, Interim Report EPRI NP-4997, December 1986.
13. EPRI Equipment Qualification Data Bank Program Description, NUS Corporation, Rev. 4, April 1987.
14. M.J. Jacobus, G.L. Zigler and L.D. Bustard, "Cable Condition Monitoring Research Activities at Sandia National Laboratories," "Proceedings of the Workshop on Power Plant Cable Condition Monitoring," Electric Power Research Institute Report EL/NP/CS-5914, July 1988.
15. T.A. Shook and J.B. Gardner, "Cable Indenter Aging Monitor," Electric Power Research Institute, Interim Report EPRI NP-5920, July 1988.
16. "Proceedings of the Workshop on Power Plant Cable Condition Monitoring," Electric Power Research Institute Report EL/NP/CS-5914, July 1988.
17. F. D. Martzloff, "A Review of Candidate Methods for Detecting Incipient Defects Due to Aging of Installed Cables in Nuclear Power Plants," Proceedings of the Workshop on Power Plant Cable Condition Monitoring, Electric Power Research Institute Report EL/NP/CS-5914, July 1988.

August 31, 1988

OPENING REMARKS

Satish K. Aggarwal
General Chairman

Good morning, ladies and gentlemen. In my opening remarks yesterday, I stated that an accident anywhere in the world is of concern to all of us. My discussions with international nuclear experts yesterday revealed that this view is shared by all without exception. Another accident anywhere in the world will threaten the nuclear option and its benefits everywhere. You must, therefore, continue to ask yourself, as I stated yesterday, "What can we do to safely operate progressively aging nuclear power plants?"

This question we should remember and try to keep always in mind. Recently Dr. Uchida, a Japanese nuclear utility official, stated to me, "All that we want is safe operation without incidents, let alone accidents. The high numbers are just results." I repeat: "All that we all want is safe operation without incidents, let alone accidents. The high numbers are just results."

This was indeed very well said. This summed up excellently my own thinking. Safety, reliability, and availability go hand in hand and are not mutually exclusive. I believe that understanding aging is the key to ensuring safety. This is the theme of this Symposium.

Yesterday, we heard several outstanding technical presentations, at times with diverse and controversial views, but I believe this is a step in the right direction.

Today, you will hear many possible actions and initiatives for the nuclear industry, but some examples on my mind include:

- Developing a good overall maintenance program for nuclear power plants
- Improving inspection, surveillance, and testing methods for safety-related equipment and systems
- Upgrading the reliability of key components, for example, trip breakers and on-site ac power sources.

I believe that insightful plant aging research programs will result in a better understanding of these objectives. Participation is needed by all in the nuclear power plant community within the United States and outside the United States. A rigorous international effort to understand plant aging is needed. I trust that this Symposium is a step in the right direction. I am sure that you will enjoy outstanding technical presentations today as well.

WELCOMING ADDRESS

Eric S. Beckjord
Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Good morning, ladies and gentlemen. I want to add my welcome to you on the second day of this International Symposium on Nuclear Power Plant Aging.

This morning, I'd like to give you a brief overview of the NRC research programs related to nuclear power plant aging, and to comment on the utilization of this research and on current rulemaking for the maintenance rule and for a license renewal rule.

Both research and rulemaking activities are integrated within the NRC Office of Research, and they will be implemented to address the technical safety issues that are related to nuclear plant aging.

I want to discuss briefly the nature of the aging process and its impact on plant safety and relate these comments to the recommendations of the National Research Council Report, which it gave two years ago on revitalizing nuclear safety research.

Commercial nuclear plants consist of many different components, systems, and structures. The designs differ and in some cases, the materials of construction differ. The plants operate in a variety of different natural and operating environments and are maintained in a variety of ways. In general, these operating and environmental conditions are both demanding and hostile, encompassing high temperature, high pressures, both sustained and cyclic operation, long-term service operation, corrosive chemical exposure and, for some components and structures, exposure to radiation.

These conditions can and do cause a time-dependent degradation of functional capabilities of safety-related systems, components, and structures. Typical mechanisms of such component degradation include material embrittlement, wear, erosion, erosion/corrosion, fatigue, environmentally assisted fatigue crack growth, and general corrosion.

Even excessive testing of components, as in the case of emergency diesel generators, can degrade component lifetime if the aging process is not managed.

A principal concern of the NRC is that plant safety could be compromised if the age degradation of plant components, systems, and structures is not detected and properly managed before a loss of function can occur.

Degradation, if unchecked, has the potential to increase risk to public safety. Therefore, we should take steps to mitigate, identify, and understand the effects of plant aging. This can be accomplished through a program of plant surveillance, condition monitoring, trend analysis, effective recordkeeping, and component and system maintenance.

Plant surveillance and condition monitoring involve the detection of incipient defects and predictive and preventive maintenance of the safety-related systems for wear, chemical reactions, cracking, and material property changes. Here, knowledge of the processes involved in material aging and of plant operations is essential. Insights on the extent of degradation can be achieved by monitoring operation and by analyzing results of the inspection and maintenance being performed. This kind of information can disclose age-related degradation mechanisms and effects that are otherwise difficult to discover in advance. The monitoring of plant performance indicators is useful in age-trending analysis. In turn, it is valuable for scheduling appropriate inspection and preventive maintenance programs that will mitigate the effects of aging.

The measure of effectiveness of a plant maintenance program is its ability to manage the effects of aging. Good maintenance will help to extend the operational life of a nuclear plant.

The National Research Council Report on NRC Research stated that there is a need for research to define the safe conditions for license extension. The report said that this research should focus on three aspects: First, long-term chemical damage to pipes, valves, and other components exposed to gases and liquids; second, long-term irradiation damage to core, structural, electrical, and instrument components; and finally, long-term effects of operational cycles on mechanical integrity.

The research developed should analyze the ability of components and systems to function beyond their design lifetime. Examples of specific research topics in this field include in situ weld repair techniques; structural integrity of plant systems, including the long-term integrity of radiation embrittled materials; on-line diagnostics to measure degradation, including nondestructive testing; and the effectiveness of in-place annealing of embrittled reactor pressure vessels.

The National Research Council Report is an important endorsement of the NRC research program on aging because many of these elements were under way in the Office of Research at that time.

Within the Office of Research, a nuclear power plant aging program has been established. The purpose of this program is to identify and resolve technical safety issues related to age degradation of electrical and mechanical systems, safety systems, support systems, and structures used in nuclear power plants. The understanding of the aging phenomena and the mitigation of these degradation effects are within the scope of the program.

The Nuclear Plant Aging Research Program, when coordinated and integrated with long-standing ongoing material and structural research programs, such as the Heavy Section Steel Technology Program and the Containment Integrity Program, provides a basis for not only determining the effects of aging, but also for establishing the technical requirements for a maintenance and life extension policy.

The goals of this integrated program are to identify and characterize those plant aging effects that could impair safety; to identify methods of inspection, surveillance, and monitoring of plants that will detect aging effects before the loss of system safety function; and to evaluate the effectiveness of maintenance and replacement practices in mitigating aging degradation.

Specifically, the program is collecting data on component aging and assessing the effects of aging on risk and reliability for normal and upset conditions. The program will provide the technical bases for license renewal, maintenance, resolution of generic issues, and revision of appropriate regulatory codes and standards. It is expected to develop methods to monitor component and system performance, to detect early signs of degradation, and to assist in evaluating the effectiveness of performance and corrective actions taken to mitigate the effects of aging.

In February of this year, the Commission directed the staff to develop a proposed rule for the maintenance of nuclear power plants. In June, the Commission further directed the staff to develop a preferred maintenance rule-making option requiring licensees to track certain performance-based maintenance indicators.

In July, a maintenance workshop was conducted in an open forum with extensive participation by the industry and by licensees. Based on the results of this work, the staff has recommended a proposed amendment to 10 CFR Part 50, the proposed rule for maintenance of nuclear power plants, which will be considered later this year.

There is a need for a maintenance program that will be effective in managing aging during the normal design life and extended life of a plant. A good program should be based on a thorough understanding of aging mechanisms. A good program should manage aging through inspection, surveillance, condition monitoring, trending, and recordkeeping of risk-significant and prioritized sets of components, systems, and structures throughout the normal operating design life.

A good maintenance program is the key to plant life extension. I believe this is the responsibility of the utilities that operate nuclear plants, and the active involvement of the nuclear industry in the elements of the NRC research program on aging will help to achieve the kind of maintenance programs that are needed for the industry.

I thank you for your attention and I hope that this Symposium will be very successful and meet your expectations.

PRINCIPAL ADDRESS

Hideo Uchida, Chairman
The Nuclear Safety Commission, Japan

THE ROLE OF MAINTENANCE IN THE MANAGEMENT OF AGING FROM JAPANESE EXPERIENCE

In Japan, verification and extension of nuclear power plant life are deemed to be important elements in our policy of further enhancing the safety of nuclear power plants now being operated. We have adopted two basic policies as the basis of dealing with aging degradation of nuclear power plants: (1) Preventive maintenance for detecting degradation symptoms of equipment at early stages and identifying proper repair and replacement plans, by conducting complete maintenance management. (2) PLEX (Plant Life Extension) Program by which the service lives of facilities/equipment of existing nuclear power plants are correctly estimated and extended by means of applying countermeasures for aging degradation and properly adopting new designs and new technologies which will be reflected in designs of the next generation of plants.

This paper presents the objectives of preventive maintenance and certain aspects of degradation phenomena, mostly on light-water reactor plants.

Present Status of Nuclear Power Development in Japan

The development of nuclear power plants in Japan was started in 1966 when a carbon dioxide gas cooled reactor (166 MWe) was introduced from the UK. Today, 35 commercial nuclear power plants are operating in Japan, with total generating capacity amounting to 27.88 GWe. This accounts for approximately 19% of the total electric power generation capacity in Japan. Most of these nuclear reactor units are light-water reactors (PWR and BWR), comprising 18 BWRs with a total capacity of 15,117 MWe and 16 PWRs with a total capacity of 12,598 MWe. A prototype heavy-water-moderated light-water-boiling reactor, named "Fugen" (165 MWe), is also in operation, and a liquid-metal-cooled fast breeder reactor, named "Monju" (280 MWe), is under construction. Taking account of nuclear power plants which are under construction, or being prepared for construction, it is expected that the total generating capacity of nuclear power plants in Japan will reach 46 GWe by the year 2000, but further efforts will still be required to attain the target of 53 GWe that is stipulated in the long range program of the Government. The electric power generation by nuclear power plants in 1987 was 185.2 TWh, accounting for 29% of the total electric power generation in Japan.

Japan has experienced approximately 340 reactor-years of operation on the nuclear power plants mentioned above by March 31, 1988. The average plant availability factor exceeded 70% consecutively for the past 5 years, being 77.1% in fiscal 1987. There has been no reactor accident nor abnormal incident which resulted in significant release of radioactive materials to the outside of the plant sites. The average frequency of automatic shutdown (scram) of reactors in operation, due to component/equipment malfunction or incorrect operation procedure, is 0.4 per reactor-year for the past 22 years, which has been reduced to less than 0.2 in recent years. (See Table 1.)

In Japan, where there are few indigenous energy resources, most of our people recognize the importance of nuclear power development, and in particular, it is recognized that light-water reactors will continue to play the major role up to the middle of the 21st century. However, it can not be denied that a large number of Japanese citizens have concerns or feelings of doubt on the safety of nuclear power. Therefore, it is all the more important to maintain and improve the levels of safety and reliability, and verify and try to extend the lives of nuclear power plants.

Fundamental Policies in Dealing with Aging Degradation

Japan has adopted two basic policies in dealing with aging degradation of nuclear power plants.

First, emphasis is placed on preventive maintenance, by which the plant maintenance efforts are thoroughly exercised to discover aging phenomena of equipment at early stages and to implement appropriate measures of repair and replacement. Namely, (1) The basic philosophy in Japan for assurance of nuclear safety lies in emphasizing creation of an atmosphere of "safety culture" among all concerned in utilities and equipment vendors as well as regulatory authorities, and recognizing the importance of such disciplinary atmosphere by all. (2) Furthermore, it should be recognized that high level technology in design, manufacture, and construction of plants; meticulous quality assurance practices; and reliable performance by operators of superior quality are important. (3) Priority is placed on prevention of abnormal occurrences and working out the necessary measures for prevention, rather than on measures for mitigating results of incidents or accidents. Personnel in charge are so trained that they constantly pay keen attention to trivial symptoms of abnormality such as leak, abrasion, corrosion, and vibration which can be perceived in normal operation or in inspection. Appropriate actions taken on such trivial symptoms lead not only to prevention of abnormal occurrences or accidents, but also to retardation and localization of aging processes.

The second policy is the PLEX (Plant Life Extension) program, for which research and development efforts are being conducted by government and private sector participants. Such efforts are divided into three broad categories. (1) Data on experiences on operation and maintenance of existing nuclear power plants, as well as on aging degradation of materials obtained by material tests and other sources, are collected and sorted to correctly assess the period of time for which relevant equipment can be properly placed in service. Identifying what is termed the critical equipment, and technology for evaluating the service lives of such equipment under actual service conditions, will be developed to predict equipment lives. (2) The technology for constantly monitoring and diagnosing aging will be developed and applied. Plant facilities, equipment, and structures, on which appropriate aging countermeasures have been applied, will be continuously monitored and will be replaced or repaired before the end of their service lives. (3) The results of these studies will be used for application of new designs, new materials, and new technologies, which will be reflected also in designs of the next generation of nuclear power plants.

As some PLEX discussions will be contributed to this symposium by Japanese participants, preventive maintenance will be mainly discussed in this presentation, with emphasis on light-water reactors.

The Japanese laws make it mandatory that a commercial nuclear power plant is subjected to inspection once a year during its plant lifetime for systems, equipment, and components relevant to safety by the Ministry of International Trade and Industry (MITI), that is the competent governmental organization having the responsibility for safety regulation through a plant's life. Refueling is normally conducted in this inspection period. The items for this periodic inspection are stipulated, as shown in Table 2, including inspection of all fuel assemblies, inservice inspection (ISI) of the primary coolant pressure boundary, and eddy current testing (ECT) on all steam generator tubes. The utility company owning the plant implements various maintenance and inspection works by its own initiative during this period of periodic inspection, for example, replacement of aged component parts, overhaul inspection of equipment, calibration of measuring instruments, and checking logical circuits, as well as repair and improvement of plant facilities when necessary. The periodic inspection usually takes approximately 3 months even if repair or improvement by the utilities is not required, which is longer than the corresponding periods in foreign nuclear power plants. However, this practice helps in enhancing plant safety and reliability, and preventing occurrence of failures, incidents, and accidents, and at the same time, symptoms of failure and aging of equipment and materials can be discovered in this process. Therefore, we believe that this periodic inspection system is an important factor in dealing with aging of nuclear power plants (Figure 1).

When construction of a commercial nuclear power plant is completed and it passes MITI's final inspection, MITI issues the operating license which authorizes the start of operation of the plant in question and permits operation of the plant through the plant's life. However, it is ruled that the plant is permitted to resume commercial operation only after the comprehensive operational test conducted at the final stage of the annual periodic inspection, confirming that the plant as a whole is qualified to the technical standard. This rule has the same effect as renewing the operating license every year, and therefore, we do not deem it necessary to introduce a system in which a safety review is conducted over again after a certain period of operation (for instance, 30 years) to permit the next period of operation. So far as engineering is concerned, facilities and components are designed and manufactured for service lives of 40 years or so, and the owner is assigned the responsibility of maintaining an appropriate quality level for facilities and equipment for the intended period. Therefore, the measures against plant aging which are being studied in Japan are based on a plant life target of 50 to 60 years, and it is intended to assess the current plant life and extend such life.

Example of Failures and Scrams in Early Stage

In the early stage of Japan's nuclear power plant development, the plant availability factors were often reduced to a 50% level, and various experiences have been gained in this period on failures. The main causes of failures experienced in early nuclear power plants were fuel cladding leak and bending, ballooning, stress corrosion cracking (SCC) in BWR plants, steam generator tube leaks in PWR plants, and other minor troubles associated with premature development (Tables 3 and 4). Most of these problems have been resolved later by research and countermeasures, except for corrosion failure of steam generator tubes that still remain in a few nuclear reactors of early construction. Leakage of fission products from fuel cladding to coolant and leakage of primary coolant from steam generator tubes to secondary coolant are seldom observed in recent years. Together with countermeasures, such as improvement of cooling water chemical treatment and employment of low cobalt materials in pressure boundary structures, these improvements have reduced the radioactivity in primary coolant to very low levels. It should be noted that such measures also have favorable effects in prevention of material corrosion and reduction of workers' radiation exposures. As stress corrosion crack and steam generator tube leak have been practically eliminated, most failures and troubles encountered in recent years are those caused by defective quality assurance in design, manufacture, and inspection, and failure in instrumentation and control sytems.

The frequency of scram due to failure in facilities and equipment of nuclear power plants was about 0.5 per reactor year in the early developing stage, but it has been reduced in recent years to a level of 0.2 per reactor year. Scram due to human error by an operator is very rare. This is deemed to be owing to education and training, refinement of manuals and observation to them by operators, and such achievements are emphasized. Although scram is the most important function for assurance of safety in nuclear reactors, unnecessary scramming would increase the potentiality of ATWS, create sudden stress changes due to thermal and pressure transients in fuels, reactor pressure vessel, pressure boundaries, etc., and can be the cause of aging degradation. Therefore, reducing scram frequency is an important issue in dealing with plant aging.

Example of Aging Degradation and Critical Equipment

SCC Issues in BWR Plants

The SCC problems which have been encountered mostly with stainless steel pipes of BWR plants have been practically resolved by changing designs to reduce excessive stress, by replacing with low carbon stainless steels, with careful water chemical treatment to reduce oxygen and chlorine contents in water, and by reducing residual stress at weldments by application of welding practice that reduces heat input and application of special heat treatment. Pipes of old BWR plant designs which were susceptible to SCC have been either mostly replaced with SCC-resistant material or heat treated at their critical weldments, thereby resolving SCC problems with such pipes.

Corrosion Damage on Steam Generator Tubes

This problem first occurred in 1972 in connection with steam generator tube leak. This phenomenon of causing tube wall thinning, which was the first problem, has been practically eliminated by changing the chemical treatment of secondary water from phosphate to hydrazine. In Japan, tube denting has not been encountered, except for cases of very small degree, since it is a recommended practice to conduct frequent blowdown of secondary coolant that contains corrosive sludges during reactor operation.

In several PWRs of early construction, corrosion defects due to SCC have been observed on steam generator tubes at U-bends, crevices between tubes and tube plate holes, expanded portion in tubes/sleeves, etc., which have been dealt with by improvement in water chemical treatment, design and manufacture, as well as by early detection of flaws and corresponding repair. Eddy current tests are applied to all steam generator tubes during periodic inspections in Japan. As a flaw extending for more than 20% of tube wall thickness is detected by this test, when a flaw is detected, that particular tube is either plugged or sleeved, even if no leakage through the tube wall is observed. This preventive maintenance practice has demonstrated magnificent improvement in performance. Of 45 steam generators equipped to 16 PWRs in Japan today, there are 10 steam generators in 4 reactors whose plugging rate exceeds 10%.

Leakage of primary coolant to secondary side through steam generator tubing has seldom been experienced due to the countermeasures stated above.

SCC Issues in PWRs

SCC has also been observed in various components of PWRs, for example, on support pins and flexible pins of control rod guide tubes. These parts have been replaced by newly designed ones.

Hair cracks have been discovered by penetrant tests on the neck of bolts (38.1 mm in diameter and 280 mm in length) that fix the flow deflection vanes of PWR primary coolant pumps. The extent of cracking depends on the duration of pump service, but they seem to occur on almost all bolts when pumps are operated for more than 50,000 hours. The cause was SCC due to inadequate selection of material and excessive stress created at the neck of bolts by too high fastening torque. It has been decided to improve the material and design, and to replace all bolts of similar PWR pumps. No serious threat to plant safety is expected even if all bolts in a pump are broken, but this is a remarkable problem of aging degradation in critical equipment. As the pump is purchased by the plant owner as a complete unit, it is expected that similar aging phenomena may occur as plants age, and particular attention should be provided on inspection of equipment that is purchased as complete units.

Neutron Irradiation Embrittlement of Reactor Pressure Vessel

It is known that the reactor pressure vessel made of low carbon steel is susceptible to embrittlement during its service life due to large doses of neutron irradiation. For this reason, specimen pieces are installed inside a vessel on the periphery of the reactor core, which can be taken out periodically to have their NDTT (nil-ductility transition temperature) measured and compared vs. their accumulated neutron irradiation measured, so that the pressure vessel life can be predicted by confirming the fracture toughness. Reactor pressure vessel materials manufactured in Japan had copper content of less than 0.2% even in those produced in the early days, and it has been confirmed by NDTT testing of surveillance pieces that such materials can stand a sufficiently long period of service. Current Japanese pressure vessel materials have copper content of less than 0.1%, and some vessels are fabricated without weldment lines on longitudinal direction. Research is also conducted on stainless steel materials used in reactor core internals by means of specimen pieces in order to investigate the neutron irradiation effect on stress corrosion cracking.

Fatigue Crack in Piping, etc.

Faults developed in piping, vessels, and equipment are often generated by, in addition to corrosion, repeated application of mechanical loads for long durations. That is, they are created by repeated stress that is caused by repetition of startup and shutdown, vibration, repeated thermal stress, creep fatigue, and fatigue caused by concentrated stress. Those repeating loads ought to have been taken into consideration in designs, but there are those which can not be anticipated in the design stage. It is important, therefore, that such flaws are detected in early stages by nondestructive tests, or the initiation of leakage is discovered, and the repair or replacement of the parts in question is implemented.

Recently, the following typical example was experienced. A rise in drain sump water level was observed in a containment of a PWR in operation. The following investigation indicated that there was water leakage of approximately 50 liters per hour on the stainless steel RHR piping (220 mm diameter, 21 mm wall thickness) upstream of the pump inlet isolation valve. This section of piping branches was from the primary coolant main piping, and water inside this piping was stagnant during normal operation. There was a crack (having opening width of 0.2 mm and 1.5 mm long) in the weldment of this piping which had been caused by fatigue due to repeated thermal stress applied to this piping, which stemmed from cyclic leaking of a small amount of primary coolant from the gland packing of the pump inlet isolation slice valve. The thermal change has been estimated to have been repeated more than 10^5 to 10^6 times. It is heard that similar examples have been found in foreign plants, and this is notable as an example of an aging problem. This particular case happened to verify the concept of LBB (leak before break), but technical study for early discovery of such hair cracks is desired anyway.

Steam Turbine

A SCC crack has been discovered on a key-way of a turbine disk boss which is shrunk fit to a rotor shaft of a low pressure turbine that has been operated for a long time in a BWR plant. Such cracks were observed on similarly designed turbine disks. Crack propagations of turbine disks are monitored to evaluate turbine lives, and they are being replaced one by one to integral (mono-block) rotors.

SCC has also occurred at blade grooves of turbine disks in PWR plants, and the groove design has been improved. Integral rotors are used in recent designs.

Application of New Technology

Progress in technology in general, as well as that related to nuclear power facilities, has been remarkable in the past 10 years. It is therefore expected to apply these new technologies for countermeasures against aging of nuclear power facilities, including adoption of new materials, improvement of welding technique, application of electronics, and enhancement of inspection methods by automatic or remote control schemes.

Conclusion

In nuclear power plants in Japan, the regulatory authority enforces strict safety regulations on each stage of design, construction, and operation of the plant, and an annual periodic inspection after commissioning. At the same time, the electric utilities take initiative in quality assurance, inspections conducted by themselves, and education and training of their personnel. Such coordinated efforts of safety assurance conducted by both government and the private sector, and the policy of emphasizing "preventive maintenance," have not only resulted in enhanced safety, operability, and reliability, but also very probably contributed to effective measures against plant aging.

Another aspect of Japan's endeavor is typified by the PLEX program which is an organized effort by government and private sectors for research and development on plant aging and plant life extension technologies.

Past examples of plant aging indicated that they were caused by SCC, corrosion, repeated fatigue, etc. Exchange of information on such relevant experiences in operation and maintenance of nuclear power plants is suggested on an international basis, because such information exchange will certainly contribute to enhancement of safety and life extension of nuclear power plants of the world.

TABLE 1. Scram Frequencies Resulting from Failures on Equipment or Components During Operation

Fiscal Year	'75	'76	'77	'78	'79	'80	'81	'82	'83	'84	'85	'86	'87
Frequencies	0.7	1.2	0.6	0.9	0.5	0.7	0.6	0.4	0.6	0.1	0.2	0.2	0.1

- Times/Reactor-Year: (Scram Times in Fiscal Year)/(Duration Hours of Electricity Generation in Fiscal Year/8,760).

TABLE 2. Scope of Periodic Inspections

BWR: *In-service inspection of reactor coolant pressure boundary, fuel assembly sipping, reactor shutdown margin,* primary coolant pump disassembly, MS safety valve disassembly, MS safety valve leakage and function, feedwater pump disassembly, CRD function, containment leakage rate test, fuel handling equipment function, etc.

PWR: *In-service inspection of reactor coolant pressure boundary, fuel assembly sipping,* primary coolant pump disassembly, steam generator heat transfer tubing, eddy current examination, pressurizer safety valve leakage, *CRD function, nuclear instrumentation function, fuel handling equipment function,* reactor containment leakage rate test, etc.

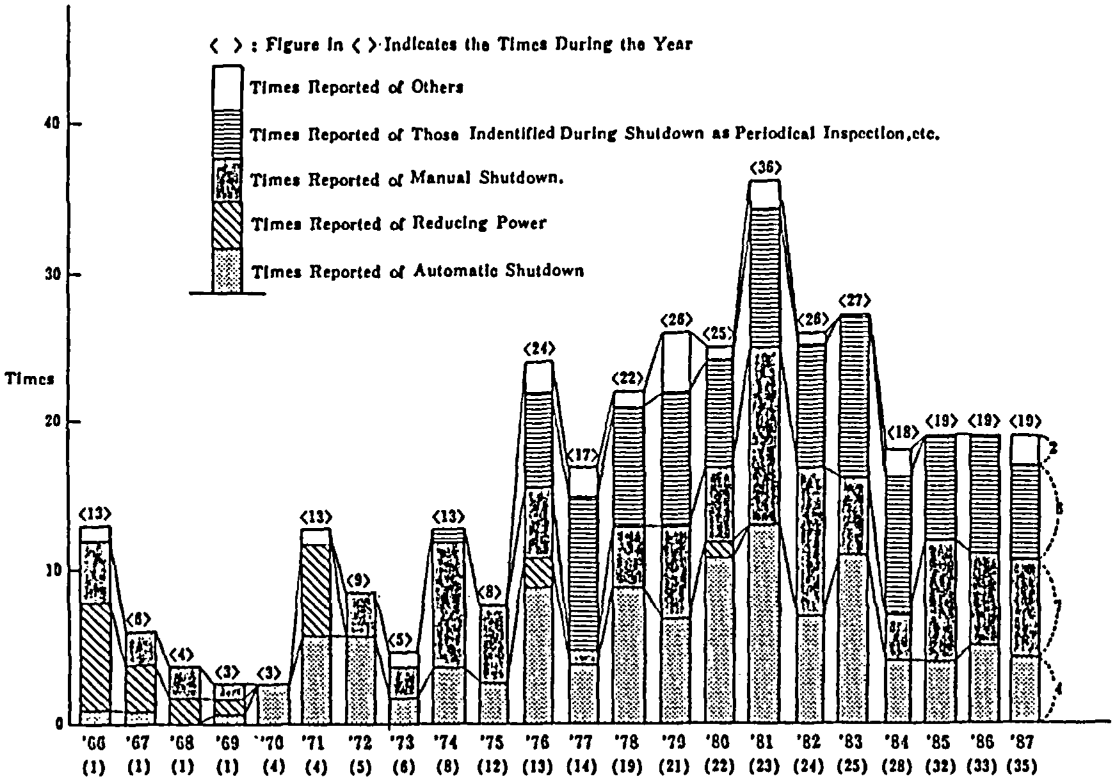
Table 3. Equipment Failure Times(Total Times up to March,1988)

System	BWR	PWR	GCR	TOTAL
Reactor System	9	21	6	36
Reactor Coolant System	32	67	9	108
Emergency Core Cooling System	3	5	0	8
Reactor Auxiliary System	4	5	0	9
Instrument and Control System	33	15	2	50
Fuel Handling System	0	1	3	4
Radioactive Waste Disposal System	3	1	1	5
Reactor Containment System	1	1	0	2
Steam Turbine System	31	9	6	46
Condensate and Feed Water System	19	8	1	28
Electrical Equipments	23	10	4	37
Common System in Electric Power Station	3	1	0	4
Ventilation and Air-Conditioning System	3	0	0	3
Auxiliary Boiler System	3	0	0	3
Others	4	1	5	10
TOTAL	171	145	37	353

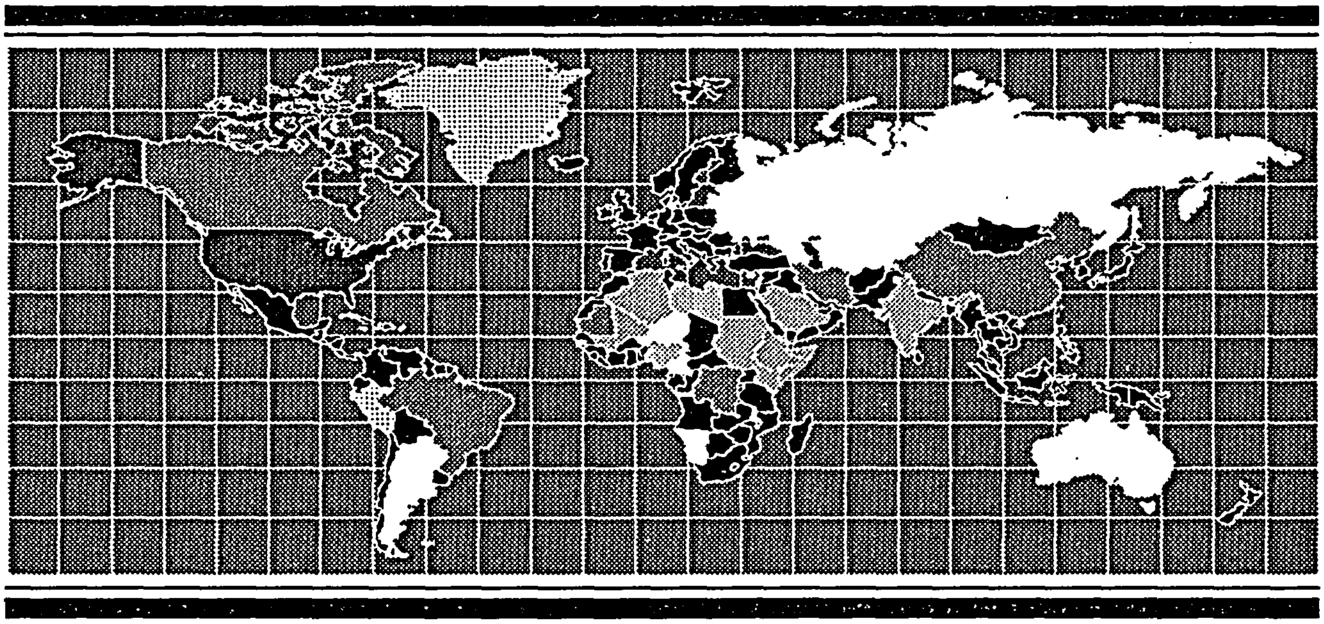
Table 4. Failure Times Categorized to Causes(Total Times up to March,1988)

	BWR	PWR	GCR	TOTAL
Unknown Causes	2	1	0	3
Incorrect Design	32	27	13	72
Incorrect Manufacture	30	42	2	74
Incorrect Construction	20	10	4	34
Incorrect Maintenance	53	15	10	78
Incorrect Operation	6	5	1	12
Incorrect Management	20	39	3	62
External Causes	7	2	0	9
Natural Aging	1	2	1	4
No Category	0	1	0	1
Others	0	1	3	4
TOTAL	171	145	37	353

Fig. 1 Accident and Incident Times



• Figure in ( ) Indicates
the Number of Reactors



TECHNICAL SESSION 4

Aging of Systems and Components

August 30, 1988

Session Chairman

L. C. SHAO
Director, Division of Engineering and Systems Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Session Co-Chairman

J. CAMPAN
Chef du Service d'Etude et de Developpement
de la Technologie des Reacteurs a Eau
CEA Center d'Etudes Nucleaires de
Cadarache, France

SAFETY ASPECTS OF NUCLEAR POWER PLANT COMPONENT AGING

Mrs M. Conte
Mr G. Deletre
Mr J. Y. Henri

Commissariat a L'Energie Atomique
France

SUMMARY

Aging, which is a natural process undergone by all materials, was identified at an early stage as one of the main phenomena liable to degrade the operating capability of the systems important to safety.

Since 1975, the methods adopted in France to manage this phenomenon are based on a policy of prevention oriented towards obtaining and preserving equipment quality. These methods, which help to delay the effects of aging, are applied at the various stages of design, construction and operation of power plants.

Minimizing or delaying the effects of aging necessitates identification of the materials sensitive to such phenomenon, as well as understanding the parameters and phenomena which govern its mechanism.

However, the latter are extremely complex and are still not completely mastered. Thus, the influence of synergies, which depend on physical parameters and on the combined action of different aging laws of components of the same item of equipment, is not fully understood.

It is on the basis of an estimate of these factors that aging tests are defined and included in the qualification procedures to which new equipment is subjected. Qualification, the purpose of which is to demonstrate the functional capability of the equipment, thus also makes it possible to check that the design meets the required quality criteria.

The tests carried out at the site at the different stages of construction, the necessary complements to the initial check, make it possible to check that the systems, after installation of the equipment, are capable of performing their functions under normal operating conditions. They also supply data which can be used to assess contingent interactions between the different items of equipment of a given system.

Finally, for safety of the installation to be guaranteed, the equipment must not age prematurely. In-service monitoring of certain parameters, indicators of aging phenomena requiring surveillance, and periodical tests supply results which are utilized to assess contingent degrading of the characteristics of the systems.

This makes it possible to adapt the frequency of preventive maintenance, the principle of which was adopted to delay aging.

It must nevertheless be borne in mind that these measures, although far reaching, are not, at the present time, in all cases representative of the state of potential degradation resulting from the combination of operating conditions in which the systems may be required.

The aging of equipment is not, at the present time, a predominant factor in the safety problems encountered in France.

Understanding of this phenomenon nevertheless remains a permanent preoccupation which justifies the current research and development actions.

INTRODUCTION

The safety of nuclear plants depends on the capacity of the systems of which they are composed to perform, throughout their planned service life and under all operating conditions allowed for in the design of the installations, the functions they were designed for.

The identification and understanding of phenomena liable to degrade this operational capacity thus constitute one of the safety problems for which allowance must be made at the earliest stage of a project.

Aging, a natural and hence unavoidable process affecting all the components of an installation, was identified at a very early stage as being one of these phenomena.

The investigation and implementation of solutions to the safety problems associated to aging make necessary:

- defining the domain in which the consequences of aging are to be evaluated,
- identifying the parameters involved,
- identifying the components sensitive to these parameters,
- understanding the mechanisms which govern its evolution.

The results of qualification tests, and of tests and checks carried out at different stages of construction and operation, as well as allowance for operating experience, constitute the necessary basis for establishing or improving the regulatory requirements. The procedures for validating components and systems of the installation are also drawn up on the basis of these tests.

Finally, the actions initiated within the scope of research and development programmes supply the additional data necessary for such validation, and provide the indispensable support for knowledge improvement.

1. SCOPE OF EVALUATION OF THE CONSEQUENCES OF AGING

1.1 Aging considered

The properties of all materials change with time. This change, which occurs at varying rates, essentially depends on the suitability of the initial properties of the material to its actual conditions of use.

Within the scope of the current studies, aging is assimilated to the evolution of the properties of safety related components during normal operation of the plants.

Operating discrepancies due to errors or to design inadequacies are not considered to be linked to aging of components.

1.2 Life-time taken into account

The operating authorizations for French nuclear plants are based on the demonstration of the safety of the installations.

The technical solutions proposed by the utility at the design stage are intended to provide safe operation of the installations for 40 years. It is this value of 40 years which was adopted for evaluating the consequences of component aging.

2. PARAMETERS CONTRIBUTING TO AGING

The parameters which contribute to aging derive from the operating conditions and from the technical solutions adopted in the design of the plant.

They can be divided into two categories:

- ambient conditions

  These are defined by the physical parameters which characterize the atmosphere prevailing in the areas where the components are located.

  Three categories of facilities are considered:

  - containment,
  - contaminatable areas,
  - non-contaminatable areas.

  The parameters considered relate to pressure, temperature, humidity, chemical composition and degree of radioactivity of the atmosphere prevailing in these areas.

- operating conditions

  These are essentially characterized by:

  - the mechanical stresses (vibration, transients and water hammer). They result from normal operation and from loading characterisics of the systems during startup or shutdown of active components, as well as interaction phenomena between components of a given system and between systems and structures of the building.
  - the stresses linked to the parameters of fluids carried by the systems (water, effluents etc.), and to fluids supplying the active components (compressed air, electric power etc.).

It is the cumulative effect of the parameters resulting from these operating conditions which will modify the initial properties of the components and thus age them. This aging must not, in particular, compromise the functional capability of the systems required under accident conditions.

The synergies of the phenomena which govern aging are extremely complex and are not yet fully understood, at the present state of art.

3. COMPONENTS SENSITIVE TO AGING

3.1 General remarks

It is customary to use the term "component" for all materials (concrete, paint, polymer, steel etc.) or equipment (cable, valves, pumps etc.) constituting an elementary system. It is in this sense the word "component" is to be understood in this paper.

3.2 Components concerned by aging studies

3.2.1 General identification of the components. The effect of the parameters contributing to aging must be evaluated for all the components necessary for the prevention of accidents and the mitigation of their consequences.

- Prevention of accidents

  For the purposes of the prevention of accidents, all components of safety-related systems in process under normal operation of the installation must be identified.

  It is obvious that all the components of the primary circuit must be taken into consideration, as must those involved in scrams, in bringing back the reactor to various safe shutdown states and in maintaining it there.

- Mitigation of consequences

  Apart from the components of the engineered safety systems of which the operating capability may be degraded by aging phenomena, the components necessary to guarantee the "radioactivity confinement" function must also be included in this study.

Finally, the list of the main system components must be supplemented with the auxiliary supporting system components necessary for their operation, as well as those of the various supporting elements and links with the containment and the concrete structures of the buildings.

In fact, preparation of such lists involves a detailed design review of each system or of each function which is safety related. This will be used to identify the active and passive mechanical components, electrical components, concrete and miscellaneous materials the degradation of which may induce the failure of a function. The system components provided for automatic or manual startup of the main and supporting auxiliary systems, or for the monitoring of their operation, must also be included in the lists.

3.2.2 Selection of components. The number of safety-related components is too high for considering to study the effect of aging on the operating capability of each of them.

It is therefore necessary to select the components which are to be studied.

The selection is carried out on the basis of design and system operating condition analyses which make it possible to classify, for each main system, the identical components (same design and same manufacturer), operating under similar ambient conditions (containment, contaminatable area or non-contaminatable area) and operating conditions (stresses).

In selecting the components, particular care must be paid to paints, varnishes, oils, greases, polymers and elastomers which are very sensitive to aging phenomena. It is frequently the degradation of the characteristics of such components which is the origin of the failure of major components such as motors, valves and pumps.

The influence of aging is studied on a component representative of the categories thus defined.

Standardization of the French nuclear units facilitates this selection and limits the number of representative components to be considered.

4. ACTION TO CONTROL AGING

Since 1975, the approach adopted in France to control aging is based on a prevention policy oriented towards obtaining and maintaining equipment quality, quality being defined as the capability of equipment to operate correctly in service. This approach, which assists in delaying the consequences of aging, is applied at the different stages of design, construction and operation of the plants.

4.1 At design stage

4.1.1 General principles. At the launch of the French nuclear power programme, regulatory requirements concerning the quality of the installation were laid down for the power plants. These are clearly stipulated in the creation permit of the installations:

"*Electricité de France must take measures to obtain for the safety-related structures, systems and components, a level of quality which is sufficient in view of the functions which they perform. An efficient system, enabling the definition of the level of quality to be attained, obtaining it, verifying the results and correcting any errors, must be implemented. This system must include the implementation of a verified series of scheduled and systematic actions, based on written and archived procedures.*" (Extract from decree 76-59 of 2nd July 1976. Creation permit of the four units of Tricastin Power Plant).

The qualification programmes, the purpose of which is to demonstrate the functional capability of the equipment, constitutes the first step of this regulatory requirement.

The general principle of the qualification sequence is shown in Table 1. Tables 2 and 3 illustrate in a more detailed manner the different tests carried out to qualify the electrical and mechanical components for normal operating conditions of the plant and, in particular, for phenomena occurring in aging mechanisms.

The "aging" tests are always carried out at the start of the qualification sequences. As an accident may occur at the end of the service life of the plant, it is essential to make sure that aging of the components does not compromise the functional capability of the safety systems and of the emergency systems necessary for managing such situations and limiting their consequences into the environment.

4.1.2 Parameters allowed for defining the aging tests. During the aging tests, damage of components must be representative of that occurring in 40 years of plant operation, the period conventionally allowed for.

Predicting the actual conditions of operation of the corresponding components in such a period includes a margin of uncertainty which it is difficult to assess. Consequently, the aging tests are established on the basis of the envelope values of the physical parameters representative of their operating conditions.

Thus, it is considered that:

- the envelope ambient conditions of an area affect all the components located in it,
- the envelope values of mechanical and thermal stress induced by a fluid are

sustained by all the components of the system carrying this fluid.

The standardization of French nuclear units and operating procedures, as well as the presence of a unique utility, also facilitate establishing these physical parameters and minimize the number of aging test sequences.

4.1.3 Aging test limits. However stringent the definition and execution of the tests may be, the tests cannot simulate actual component aging.

The representativeness of the tests is particularly limited as:

- in the test facilities, a complete system cannot be checked and not all the physical parameters governing the aging phenomena can be taken into account at the same time. The tests are carried out sequentially and the synergy of the phenomena is thus only partially allowed for,
- the duration of the tests may be too long when it is necessary to extrapolate the acceleration laws up to the technological limits of certain components.

Despite this limitation, the lessons drawn from such aging tests are of fundamental importance.

In particular, they make it possible:

- to monitor, at different stages of accelerated aging, the properties of components and identify those which are more sensitive to particular phenomena,
- to constitute a database concerning the foreseeable behaviour of components, the data can then be used together with that obtained during operation of installations to prepare component maintenance systems,
- to eliminate any components which are not suited to the planned conditions of operation.

If, for a given function, there is no component which has passed the aging tests, a request for a concession may be made by the utility. Analysis of the proposals frequently leads to reducing the duration of use of the component to a value compatible with maintaining its functional capability. If necessary, a research and development programme is initiated.

Thus, the presence of components such as seals, insulators, plastic and grease has resulted in limiting the operating period of isolation and adjustment valves pneumatically actuated. 25 valves of this type are used in each French PWR, 12 of which provide the "containment isolation" function. The operability of the most sensitive component, the diaphragm used in the pneumatic actuator, was only guaranteed for an integrated dose of 100 kGy. Therefore, the utility has proposed, for the qualification tests, an integrated dose level compatible with such guarantee:

- Aging irradiation level 35 kGy,
- Accident irradiation level 65 kGy.

In complement of this proposal, the utility commits itself to replace, every ten years, the component sensitive to the aging due to irradiation.

In order to evaluate the validity of the utility proposals, the maximum values of the corresponding doses actually integrated by the diaphragms have been estimated on realistic basis; then the potential consequences of the valve failure have been evaluated.

As the safety analysis has demonstrated the compatibility of these proposals with the safe operation of the plants, the concession has been accepted. Furthermore, in order to maintain sufficient safety margins, the utility commits itself to replace the diaphragms every five years.

4.2 At construction stage

The tests carried out at different stages of construction of the installation constitute the first tests demonstrating the functional capability of the components and systems under operating conditions which are very similar to those occurring during operation. In particular, the synergies associated with the interaction between the components and the interactions between systems and structures are present during such tests. The functional values and parameters specific to the components, recorded under these conditions, are thus characteristic of the state of the installation at the start of operation. They make it possible to supplement the database for aging tests and constitute the "zero point" from which evolution in functional capability due to aging will be monitored.

4.3 At operation stage

The restarting tests, the periodical tests and the in-service surveillance constitute the principal means of following up of aging. They also make it possible to increase the understanding of the phenomenon.

The action initiated to control aging is based on the results of these tests and on surveillance operations.

5. CONTROL OF AGING

5.1 Principles

The mechanisms which govern aging are not yet fully identified, and control of this phenomenon remains difficult.

The action initiated is intended to limit or delay the consequences of aging of components for the safety of the installations. It is based on analysis of changes in the criteria and performance levels of components and systems as observed during operation of the plants, and consist in replacing or repairing the degraded components.

Although the French nuclear power plants are still relatively young, the older plants yet attained 70 000 operating hours. Current availability of the installations has shown the high quality of operation. Maintaining this performance level will depend greatly on the quality of the maintenance work planned, particularly for the components sensitive to aging phenomena.

5.2 Maintenance

Different types of maintenance are applied on French installations:

5.2.1 curative maintenance: this consists in replacing or repairing faulty equipment,

5.2.2 preventive maintenance: this consists of scheduling maintenance work on the basis of criteria relating to operating hours or a numbers of startups.

The preventive maintenance programmes include following up of the behaviour, under the operating conditions and, in particular, under irradiation, of the lubricants necessary for preserving the functional capabilities of the systems, as well as their replacement.

The aging laws of these components are not well understood, specific tests have therefore been carried out on simulation testing benches to both select the lubricants best suited to the requirements and also to optimize the frequency of their replacement.

These tests have already made it possible to select greases adapted to operation of the emergency safety system pumps and of the residual heat removal system. Other tests are in progress to identify the lubricants best suited to the operation of other pumps.

5.2.3 predictive maintenance: here, the maintenance work is initiated on the basis of physical criteria representative of the state of the equipment in operation. This maintenance, which is considerably more sophisticated, is based on in-service surveillance of the installations.

At the present state of art, predictive maintenance cannot be applied to all the components sensitive to aging. Nevertheless, preventive and predictive maintenance already represent 70% of the activity carried out in French installations in this field.

Generalizing the predictive maintenance necessitates identification of all the physical parameters which are the most sensitive to aging phenomena, and development of the necessary methods for following up their progress.

Research and development programmes are under way to meet these requirements.

CONCLUSIONS

Maintaining the quality of the installations, which is necessary to ensure a sufficient degree of safety and availability of the plants, depends on the development of maintenance activities.

The aging of components must be limited or delayed to guarantee that the criteria provided for at the design stage are met throughout the planned service life of the installation.

Should plant life extension be envisaged, the actions and arrangements implemented to control the aging of components will contribute to finding the solutions necessary in attaining this new objective, which can be only reached if the safety of the plants remains proved.

TABLE 1

GENERAL PRINCIPLES OF MECHANICAL AND ELECTRICAL EQUIPMENT QUALIFICATION

LOCATION	NORMAL OPERATION				ACCIDENT PHASE			POST ACCIDENT	EQUIPMENT CLASSIFICATION	
	Aging									
	Thermal	Functional	Irradiation	Mechanical vibrations	Earthquake	Irradiation	Thermodynamic conditions		Electrical	Mechanical
	V1	V2								
	V		I	5		1	A	P.A.		
INSIDE CONTAINMENT	X	X	X	X	X	X	X	X	K1	
				X	X					M1 Valve: electrically actuated
	X	X	X	X	X	X	X	X		M1 Valve: electricopneumatically actuated
OUTSIDE CONTAINMENT		X		X	X				K3	
				X	X					M2 Valve: electrically or electropneumatically actuated

TABLE 2

TEST PROCEDURE FOR IN CONTAINMENT ELECTRICAL EQUIPMENT

Preparation of procedure and reference tests

- Examination of identification dossier and supplementary documents
- Tests of dielectric strength and/or isolation resistance measurement
- Measurement of functional characteristics

Tests at functional operating limits of equipment

- Linked to nature of equipment
- Linked to equipment installation conditions (ambient pressure, ambient temperature etc.)

Assessment of equipment behaviour over a period of time

- Thermal aging test
- Accelerated damp heat test
- Prolonged operating test
- Cumulated radiation during operation
- Mechanical vibrations resistance test

Stresses due to accident conditions

- Earthquake resistance test
- Cumulated irradiation during an accident inside the containment
- Thermodynamic and chemical conditions representing an accident inside the containment

TABLE 3

QUALIFICATION OF ELECTRICAL EQUIPMENT ACCELERATED AGING

Tests for assessment of equipment behaviour over a period of time

- Thermal aging tests
- Dry heat tests
- Cold tests
- Sudden temperature change tests
- Resistance to mechanical vibrations
- Resistance to mechanical shock
- Accelerated damp heat tests
- Cyclic damp heat tests
- Saline mist tests
- Dust penetration tests
- Cumulated irradiation during operation of reactor under normal conditions
- Prolonged operating tests

RESULTS OF AN AGING-RELATED FAILURE SURVEY OF LIGHT WATER SAFETY SYSTEMS AND COMPONENTS[a]

Babette M. Meale, David G. Satterwhite, Philip E. MacDonald
Idaho National Engineering Laboratory

ABSTRACT

The collection and evaluation of operating experience data are necessary in determining the effects of aging on the safety of operating nuclear plants. This paper presents the final results of a two-year research effort evaluating aging impacts on components in light water reactor systems. This research was performed as a part of the Nuclear Plant Aging Research program, sponsored by the U.S. Nuclear Regulatory Commission.

Two unique types of data analyses were performed. In the first, an aging-survey study, aging-related failure data for fifteen light water reactor systems were obtained from the Nuclear Plant Reliability Data System (NPRDS). These included safety, support, and power conversion systems. A computerized sort of these records classified each record into one of five generic categories, based on the utility's choice of the failure's NPRDS cause category. Systems and components within the systems that were most affected by aging were identified. In the second analysis, information on aging-related reported causes of failures was evaluated for component failures reported to NPRDS for auxiliary feedwater, high pressure injection, service water, and Class 1E electrical power distribution systems.

1. INTRODUCTION

Time or cyclic-dependent degradation (aging) mechanisms, such as wear, corrosion, and fatigue, have caused component failures. These failures have raised concerns about the effectiveness of the safety equipment at some operating plants. Many of these aging-related issues have been and are being addressed by the nuclear industry through research, improved designs, standards development, and especially, improved operation and maintenance practices. Nevertheless, the aging degradation of certain components will continue, and as the light water reactor (LWR) population ages, currently unrecognized degradation effects are likely to arise.

This paper summarizes the results of two different and unique NPRDS data examination and manipulation studies concerned with the identification of LWR safety systems and components that have been significantly affected by aging phenomena and the identification of the specific aging failure causes for selected systems and components. These analyses were performed as part of the U.S. Nuclear Regulatory Commission research effort for the Nuclear Plant Aging Research (NPAR) Program. Complete documentation of these two analyses is contained in Reference 1.

The first task was an aging-survey analysis, which consisted of a computerized sort of approximately 17,500 NPRDS component failure records from fifteen different LWR safety and support systems, including systems used in both pressurized water and boiling water reactors. The purpose of this survey was to identify systems and associated components most affected by aging. NPRDS failure records are classified by utility personnel into nine categories referred to as cause categories. This analysis utilized the cause categories to systematically classify (using computerized sorting techniques) NPRDS failures into five generic failure categories. These generic failure categories are aging, design and installation error, testing and maintenance activities, human-related actions, and other (cause unknown or unclassifiable) failures. Mechanisms of failure were not identified during this analysis since mechanism determinations require detailed information concerning the component failure. This type of information is not available from the coding structure utilized by the NPRDS. The resulting data of the analysis allowed a determination of the LWR safety systems that have been significantly affected by aging mechanisms, as reported by the utilities to the NPRDS. The data also allowed the relative magnitude of aging effects between systems to be evaluated.

The second task was an aging-related reported cause of failure analysis. This analysis consisted of meticulous examination of selected NPRDS failure records. A failure-cause determination and aging classification, to the level of resolution available in the NPRDS failure reports, were assigned to component failures in four safety and support systems. The resulting failure-cause data were analyzed to identify the dominant reported aging mechanisms and the components most affected. This information is useful in the evaluation of the influence of aging on plant risk using probabilistic risk assessment (PRA) techniques. In this application, the absolute magnitude of the aging effects is not required. Relative impacts are useful for the modification of existing PRA failure rate data.

1.1 Aging Definition and Classification

The failures determined to be aging related in the aging-survey analysis were the failures that the utility personnel had categorized under the NPRDS wearout category. The NPRDS guideline for classifying a failure under their wearout cause category is any failure thought to be the consequence of expected wear or aging.[2]

The definition of aging used in the reported cause of failure analysis is:

> Aging is the degradation of a component, resulting in loss of function or reduced performance caused by a time-dependent agent or mechanism. The agent or mechanism can be cyclic (e.g., caused by repeated demand), or continuously acting (e.g., caused by the operational environment). The change in component failure probability resulting from degradation will

---

[a] Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570.

monotonically increase with the time of exposure to the agent or mechanism unless the component is refurbished, repaired, or replaced.

Different types of aging agents or mechanisms affect a component during its operational life. Environmental effects, like normal wear of component parts, erosion, corrosion, cyclic fatigue, etc., tend to affect the component in a continuous fashion at rather low aging rates. Other types of aging stem from activities affecting the component through a random event. An example of the latter type is a random maintenance error causing the component to experience significantly accelerated aging through a mechanism such as binding, resulting in wear. Random maintenance errors are usually not identified in a failure report. There are cases where a mechanism is identified as causing a failure but depending upon the circumstances under which the failure occurred, the mechanism could be considered aging or random. Therefore, the practical application of this definition leads to some uncertainty due to difficulty in distinguishing between aging-related failures and random failures solely on the basis of a failure description or reported failure causes. Engineering evaluations (materials, stressors, environment) of the failed components and knowledge of the component maintenance histories are sometimes necessary to accurately identify aging-related failures.

1.2 Failure Event Data Source

The component failure data selected for analysis were obtained from the Nuclear Plant Reliability Data System (NPRDS), a component failure data system owned by the Institute of Nuclear Power Operations (INPO); INPO member utilities provide data to the NPRDS on a voluntary basis. The nuclear power plant-specific NPRDS data are proprietary. Therefore, the data presented in this paper have been made generic so events cannot be traced to specific plants or component manufacturers.

The NPRDS data source has several strengths and limitations for aging evaluations of plant safety systems, support systems, and components that reflect on the quality of the data and its applicability for certain uses and interpretations. These strengths and limitations are itemized in Reference 1. In summary, NPRDS is advantageous to aging research in that it contains an extensive amount of component data from multiple utilities that are not available from other sources. NPRDS also provides detailed information concerning the component and its failure. However, NPRDS does not provide complete maintenance histories of the failed components. Therefore, time-line histories needed for aging evaluations are not obtainable. Also, accurate component service age calculations are difficult to obtain from NPRDS because the in-service date does not reflect preoperational testing or component refurbishment. Furthermore, plant-specific effects such as maintenance, environment, and other factors are masked when interpreting NPRDS data aggregated over all reporting utilities.

Therefore, NPRDS data can supply only relative information regarding which LWR safety systems and components have been significantly affected by aging and the underlying causes of that aging. Accurate determinations require analysis of plant records, which is beyond the scope of this study. However, for aging evaluations that rely on PRA techniques, only relative information is needed to modify the existing PRA information.

2. AGING SURVEY RESULTS

In the aging-survey analysis, approximately 17,500 NPRDS failure records were categorized to determine system and component level aging effects. A computerized sort classified component failures into five generic failure categories, based on the utility assigned NPRDS cause category. The correspondence between the NPRDS cause categories and the five generic failure categories is illustrated in Figure 1. As indicated in Figure 1, the only NPRDS cause category corresponding to the generic category of aging is wearout.

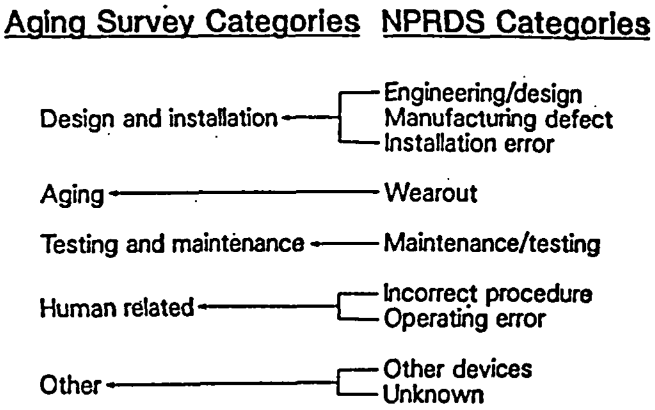


Figure 1. Relationship of NPRDS cause categories and the aging-survey failure categories.

NPRDS failure data were compiled for pressurized water reactor and boiling water reactor systems and their major subsystems. The vendors represented in this survey were Westinghouse Electric Corporation (WEC), Babcock & Wilcox Company (B&W), and General Electric Company (GE). It is recognized that several of these systems were designed by an architect/engineering firm and are not reactor vendor specific. However, the NPRDS is structured to supply system information by reactor vendor only. The systems surveyed and their associated vendors and total failure count are presented in Table 1.

Detailed data resulting from this computerized survey of the NPRDS are contained in Reference 1. Major results obtained during the survey analysis are summarized in this paper. The categorized data were aggregated with respect to several different boundaries such as all component failures within a system, all failures of a particular component within a system, or all failures of a system-specific component during a particular age interval. The results are presented in terms of failure fractions with respect to these boundaries. Failure fractions are used as a comparative measure of how important a certain category, such as aging, is within particular boundaries.

Failure fractions were calculated for the five generic failure categories and the five NPRDS system-effect categories. These fractions represent an aggregation of all components within a system. Failure fractions within a particular system were calculated by dividing the total count for a failure category by the total failure count for that system. System-effect fractions were calculated in a similar manner. Results of these two analyses are illustrated in Figures 2 and 3. Data for the five generic failure categories are illustrated in Figure 2 and data for the five NPRDS system-effect categories are illustrated in Figure 3.

TABLE 1. AGING SURVEY SYSTEMS

System	Acronym	Vendor	Failure Count
Class 1E electrical power distribution	1E	WEC, B&W, GE	2259
Auxiliary feedwater	AFW	WEC, B&W	829
Component cooling water	CCW	WEC, B&W, GE	976
Containment fan	CTF	WEC	318
Containment isolation	CTIS	WEC, B&W	926
High-pressure injection	HPIS	WEC, GE, B&W	2029
Low-pressure injection	LPIS	GE	458
Main feedwater	MFW	B&W, GE	827
Reactor building cooling	RBC	B&W	65
Reactor core injection cooling	RCIC	GE	774
Reactor protection trip	RPS	WEC, GE, B&W	3533
Reactor coolant	RXC	GE, B&W	1012
Residual heat removal	RHR	WEC, GE, B&W	1968
Service water	SWS	WEC, B&W, GE	1272
Standby liquid control	SBL	GE	173

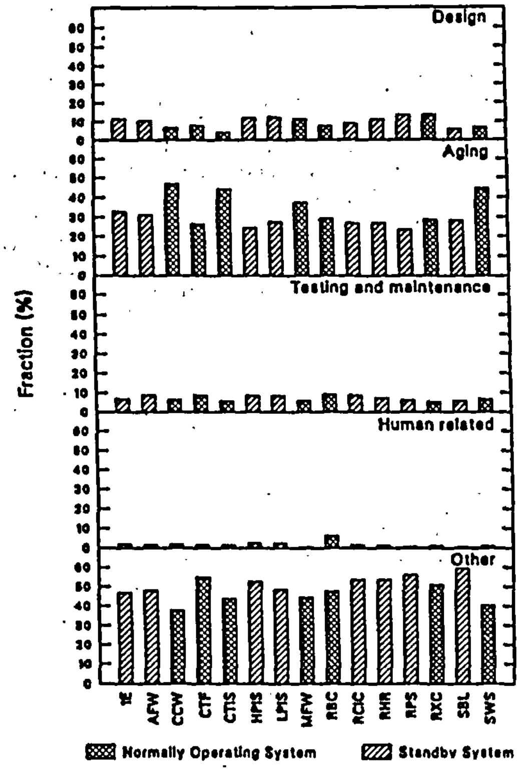


Figure 2. Relationship of failure category fractions by systems.

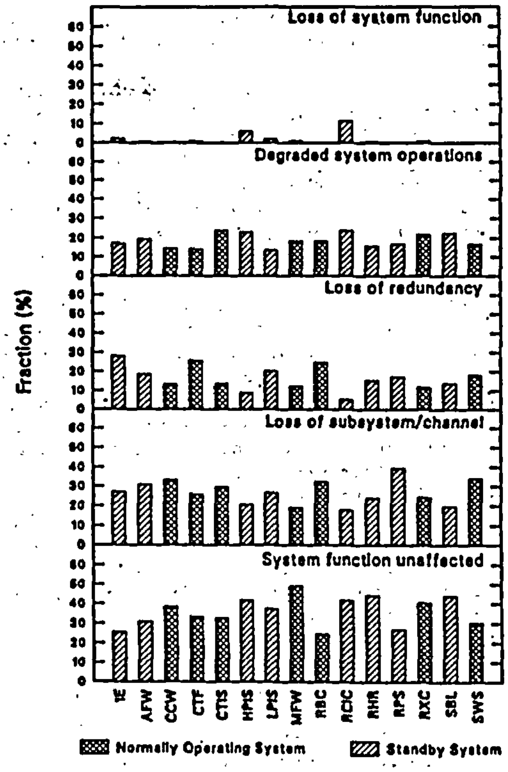


Figure 3. Relationship of system-effect category fractions by systems.

The data presented in Figure 2 indicate that the dominant failure categories are aging and other. Aging effects contributed to approximately 32% of all the failures reported; approximately 49% of the failures reported were categorized as other. Human-related failures only contribute about 1.5% to the total of failures reported. Failures in the testing and maintenance and design and installation categories account for approximately 7.5% and 10% of the total failures, respectively.

The other category consists of failures whose cause could not be identified by utility personnel or could not be assigned to another NPRDS category. The number of failures in the other category is indicative of the difficulties encountered in determining the cause of failure for certain components and the practice of replacing a component or piece part without establishing the reason for failure.

Examination of aging failure-category data indicates that, in general, normally operating systems exhibit slightly higher aging fractions than standby systems. The average aging fraction for normally operating systems is 36.8%, while the average aging fraction for standby systems is 27.7%. The normally operating systems with the highest aging fractions are component cooling water, service water, and containment isolation. The component cooling water system in nuclear power plants is operating at all times. This constant operation results in a higher incidence of aging failures. River, lake, and ocean water are normal sources of service water. This influent water contains particulate and debris that accelerate some aging mechanisms such as erosion and corrosion, and increase the chance of failures due to foreign material intrusion. Containment isolation components must meet strict criteria defined in the technical specifications or are considered failed.

The relationship of failures to the reported system effects is presented in Figure 2. Examination of this figure indicates that although the fractions in the system function unaffected category are slightly higher than the other four system-effect categories, there was no clearly dominant system-effect category. However, very few failures caused a total loss of system function in any of the systems.

The impact of the failure categories on the performance of a particular component within a specific system can be evaluated at both the component level and system level. For the component-level fractions, the total failure count per failure category for a particular component are divided by the total failure count for that component within the appropriate system. This comparison is useful in determining the aging impact on the performance of the component. In the system-level component failure-category fractions, the numerator remained the same but the denominator for the fractions was the total failure count for that system. This comparison gives a representation of the failure category's importance within the system. However, this should not be confused with an importance that would be obtained from a probabilistic risk assessment calculation.

To measure the uncertainty of the aging-fraction data, a statistical uncertainty study was performed on the component-level fractions for the system-specific component aging fractions (denominator of which is total number of failures for the particular component within its respective system) to identify statistically the components that exhibit a system-dependent aging impact. Most components exhibited no statistical dependency related to aging. Only four components demonstrated any statistically significant differences between aging fractions across systems. These components were valves in the containment isolation system, pumps in the component cooling water system, supports in the high pressure injection system, and switches in the Class 1E electrical power distribution and reactor protection systems (see data in Table 2).

Since valves and pumps exhibited system dependency and are also significant contributors to risk in probabilistic risk assessment models (supports and switches are not generally modeled), the following discussion will focus on the results of the analyses performed on the data for these two components. Findings based on the data of other selected components are documented in Reference 1.

Aging results for these two components are discussed with the aid of two sets of figures. The first figure of each set displays the component aging fractions at the component level (e.g., the number of aging-related failures divided by the total number of failures of that particular component in a specified system) along with their confidence intervals. The numbers shown with each data point are the number of failures associated with that component in that system. The second figure in each set presents the component-aging fractions at the system level, which reflects the importance of aging for that component within the particular system.

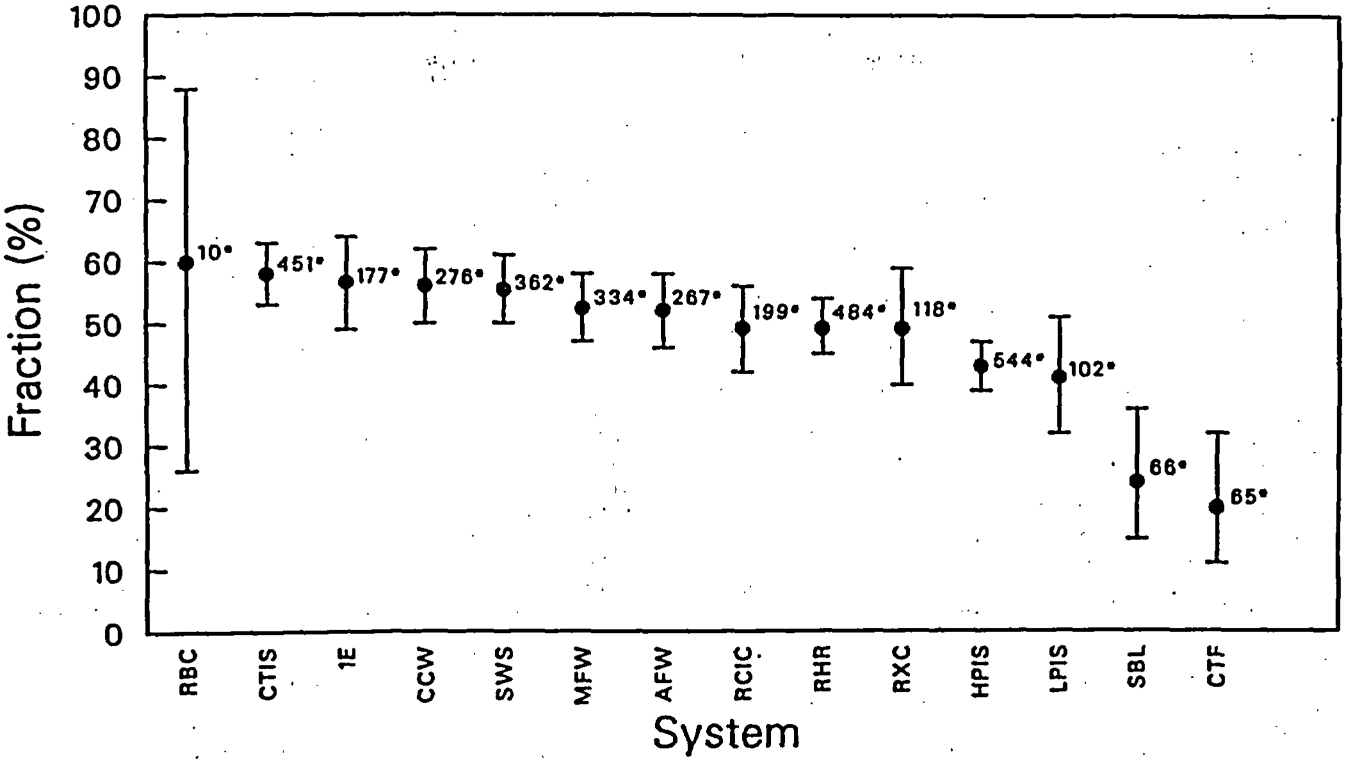
Aging-fraction data for valves are presented in Figures 4 and 5. The data presented in Figure 4 indicate that component-aging fractions for valves are relatively significant and consistent between systems at 50% to 60% with the exception of standby liquid control and containment fan systems, where valve aging failures were less evident. Examination of Figure 4 reveals that the associated confidence intervals for the valve fractions are reasonably small (20%) due to the large valve failure populations. The confidence intervals imply that aging in valves is independent of the system in which the valves reside. However, the uncertainty study results indicate that aging in valves residing in the containment isolation system is system dependent. The uncertainty study results state that valves in the containment isolation system have a higher aging fraction (58%) than other systems (49% composite) containing valves. Even though analyses of the aging fractions of valves at the component level indicate that there is little system to system variability, the data presented in Figure 5 indicate that at the system level, systems can be identified where aging of valves has a greater impact on system performance. The data presented in Figure 5 indicate that systems where the impact of valve aging is relatively important are containment isolation (28%) and main feedwater (21%) systems, followed by auxiliary feedwater (17%), component cooling water (16%), and service water (16%) to a lesser extent. Also, the data in Figure 5 indicate that at the system level, the average aging fraction for valves is 12.5%. This is the highest average aging fraction at the system level for any of the NPRDS component designations.

TABLE 2. UNCERTAINTY STUDY: STATISTICAL SYSTEM DEPENDENCY RELATED TO AGING

Component	Aging Fraction	95% Confidence Interval	Systems
Instrumentation-switch[a]	0.19	(0.16, 0.23)	1E, RPS
	0.10	(0.09, 0.13)	HPIS, LPIS, RCIC, RHR, RXC, SWS
Pump[a]	0.75	(0.70, 0.81)	CCW
	0.45	(0.42, 0.49)	1E, AFW, HPIS, LPIS MFW, RHR, RXC, SBL, SWS
Supports[a]	0.37	(0.27, 0.48)	HPIS
	0.20	(0.16, 0.24)	LPIS, MFW, RCIC, RHR, RXC
Valve[a]	0.58	(0.53, 0.63)	CTIS
	0.49	(0.47, 0.51)	1E, AFW, CCW, CTF, HPIS, LPIS, MFW, RCIC, RHR, RXC, SBL, SWS

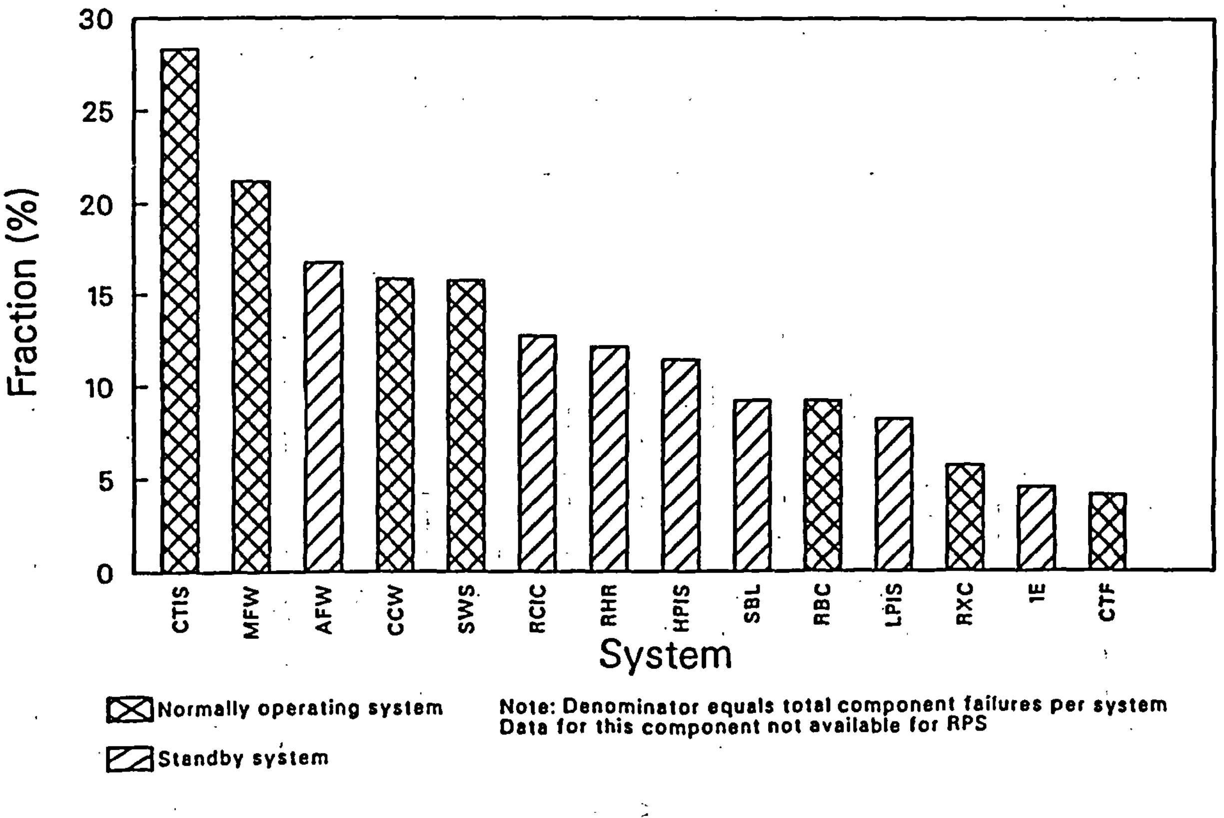
---

a. Systems with expected cell count of less than 5.0 were deleted. The remaining system-specific component data were used to calculate the aging fraction and confidence interval.



* Total valve failures per system (denominator)
Data for this component not available for RPS

Figure 4. Fractions of valve failures for specified systems due to aging.



Note: Denominator equals total component failures per system
Data for this component not available for RPS

Figure 5. Fractions of total component failures for specified systems due to aging of valves.

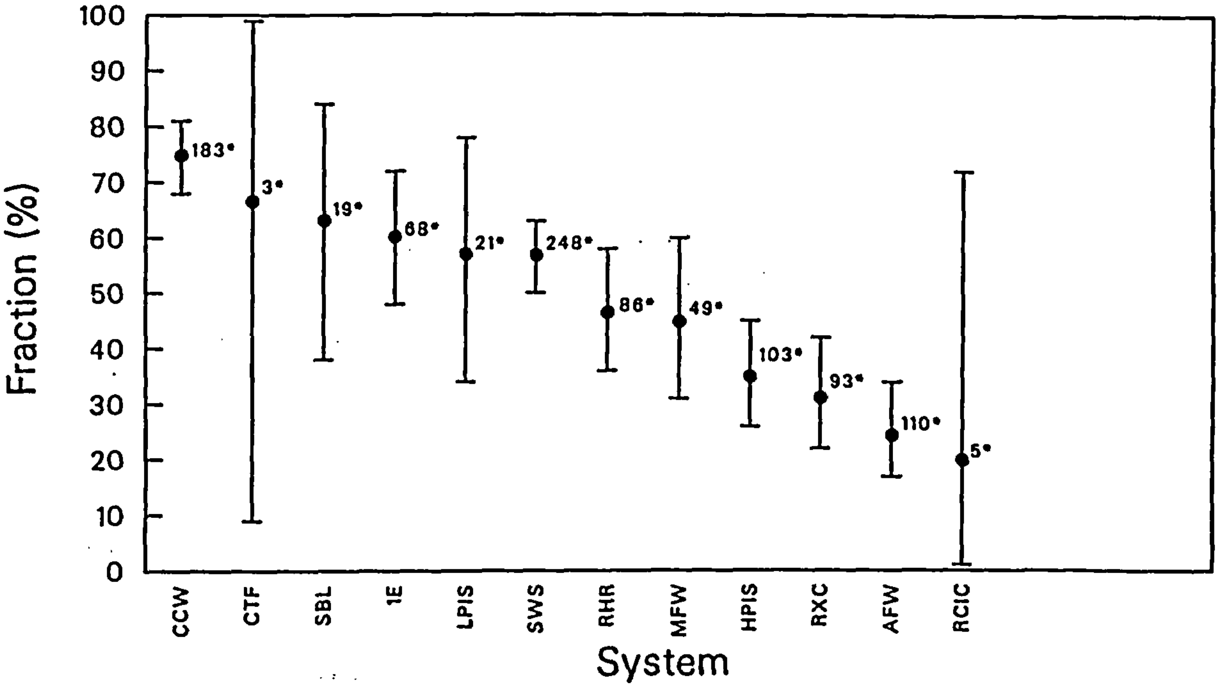
Therefore, the aging-related failure of valves is important in the systems in which they reside, when compared to other components.

The aging-fraction data for pumps are presented in Figures 6 and 7. The data presented in these figures identify pumps in the component cooling water system as having the highest aging fraction at both the component level (75%) and the system level (14%). This is a reasonable observation since the component cooling water system in a nuclear power plant is operating at all times. The constant pump operation results in increased aging effects. This is a unique factor since most systems within a nuclear power plant are not operating during all phases of reactor operations. Some systems are not required during refueling outages and some are placed in a standby mode during power operations. The uncertainty study further indicates that the component cooling water pumps have statistically distinguishable aging impacts when compared to other systems. The uncertainty study results state that component cooling water pumps have a higher aging fraction (75%) than pumps in the other eleven systems containing pumps (45% composite). These aging fractions indicate that pump aging-related failures have a significant impact on the operation of their systems. The data presented in Figure 6 also indicate that the confidence intervals associated with the pump-aging fractions vary considerably and are larger than the ones associated with valves. Additionally, the data presented in Figure 7 indicate that, at the system level, aging in pumps within the service water (11%) and standby liquid control (7%) systems is slightly more important than aging in pumps within the other nine systems containing pumps, all of which exhibited less than 5% aging-related impact. Pumps in both of these systems are adversely affected by the fluid which they pump. Service water is generally river or ocean water, which contains particulates and debris that plug, erode, and corrode the pump internals. Standby liquid control is a boric acid injection system. The boric acid environment can be very damaging to pump seals.

A study was done on the time dependency of aging-related failures for system specific components. The time-dependent fractions for valves in six different systems along with their estimated confidence intervals are illustrated in Figure 8 and pump data are illustrated in Figure 9. The systems chosen for the illustration were the six systems with the largest component populations for the particular component. The fractions were calculated using the component failure population within each of the time intervals (number in parentheses in the figures) as the normalizing basis.

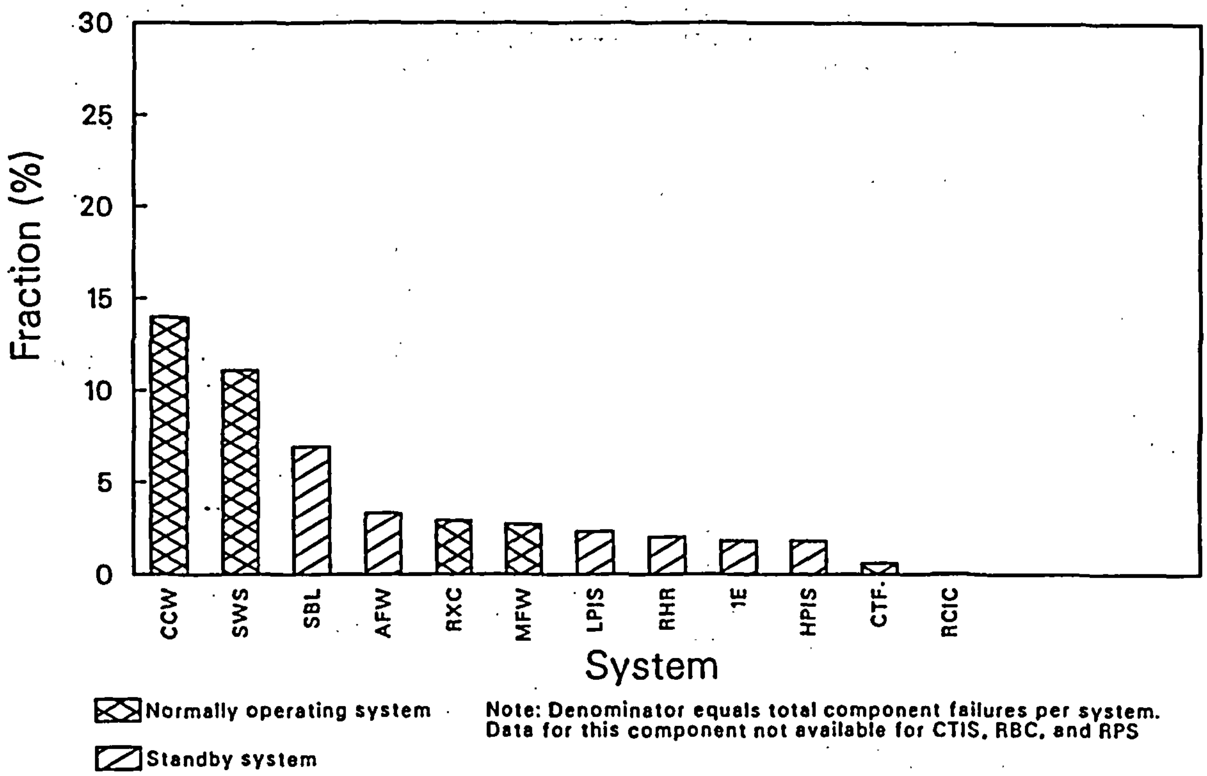
It was originally expected that this study would find an increase in aging failures with an increase in the age of the component up to the useful lifetime of the component. However, some of the components studied may have a useful life of only 5 to 10 yr and the useful life of some of their replaceable piece parts may be significantly shorter. This is especially true of the component piece parts or internals that age more rapidly than the component structural parts. Furthermore, very few components have experienced an exposure time of 15 to 20 yr. This causes the existing time-dependent aging fractions to be very uncertain due to sparse data. This is illustrated in Figures 8 and 9 by the large confidence intervals calculated for the fractions in the 15- to 20-yr division. Additionally, the data utilized in this study are affected by variables such as plant maintenance practices, the age of the plant, and NPRDS reporting practices. Plant maintenance often results in the complete rejuvenation of a component; however, occasionally, plant maintenance results in accelerated component degradation.

Data exhibited potential time-dependent trends when the postulated confidence intervals are considered, with most components having an increasing (to some degree) failure rate with time (see the data for valves in the high-pressure injection, residual heat removal, and main feedwater systems in Figure 8). However, some components may statistically or actually



* Total pump failures per system (denominator)
Data for this component not available for CTIS, RBC, and RPS

Figure 6. Fractions of pump failures for specified systems due to aging.



Note: Denominator equals total component failures per system. Data for this component not available for CTIS, RBC, and RPS

Figure 7. Fractions of total component failures for specified systems due to aging of pumps.

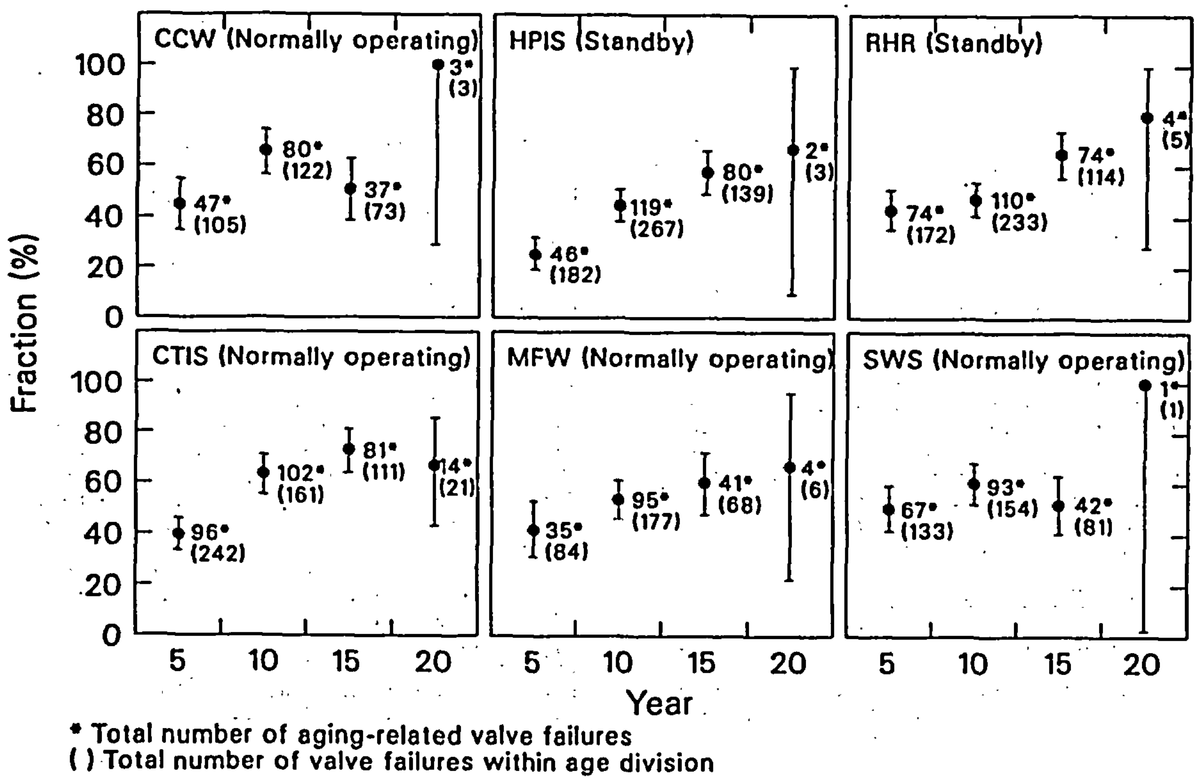


Figure 8. Time-dependent aging fractions for valves.

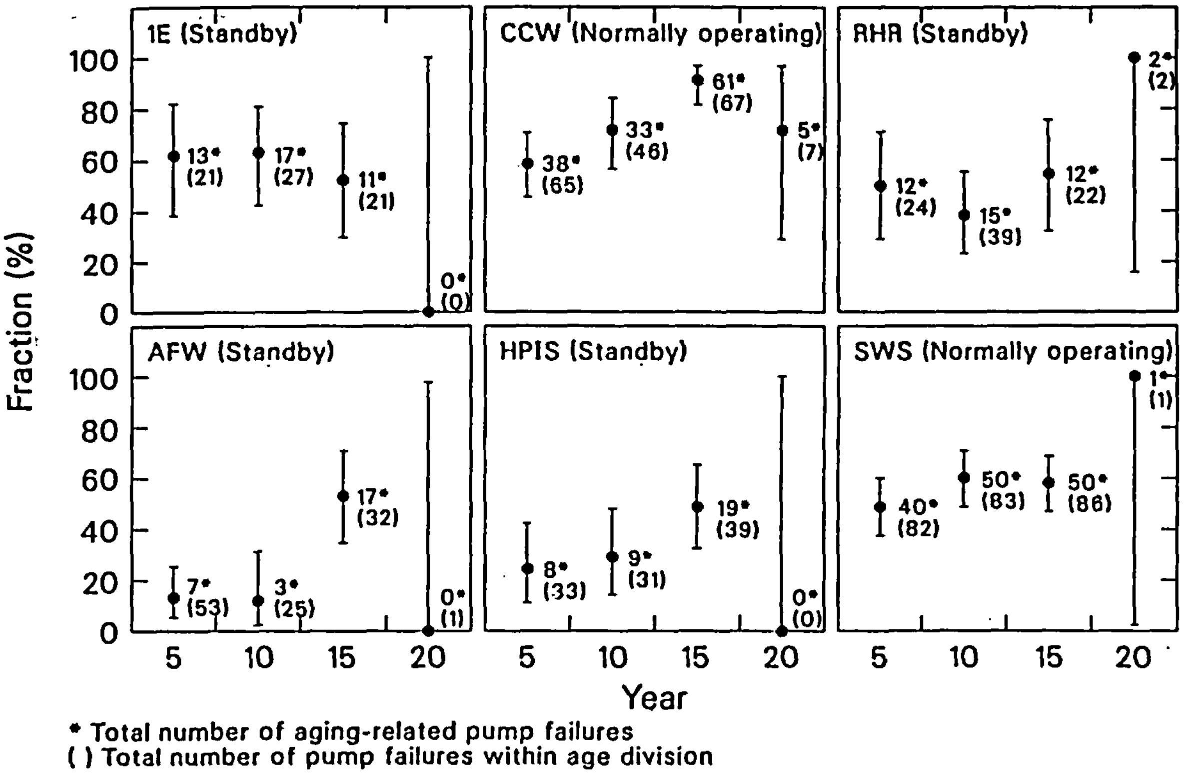


Figure 9. Time-dependent aging fractions for pumps.

have constant failure rates, which would indicate that aging effects are not important in these components over their lifetimes (see the data for pumps in the Class 1E electrical power distribution system in Figure 9 and the data for valves in the component cooling water system in Figure 8). Statistical analysis could be used to calculate the failure rate increases but such an analysis was beyond the scope of this study.

Based on the analyses in this study, the components that exhibit statistical system dependency and significant aging fractions are valves and pumps. A ranking of the component-aging fractions, at the system level, for all the components within the various systems further supports this observation. This ranking is illustrated in Figure 10 for all system-specific components which have system-level aging fractions greater than 5%. As indicated in Figure 10, valves in the containment isolation system exhibit the highest aging-related fraction (28%) of all the components analyzed. Valves in main feedwater, auxiliary feedwater, component cooling water, and service water are also among the top five components in this ranking. Pumps in the component cooling water system exhibit the highest aging-related fraction for pumps at the system level and rank sixth on the list of important components.

3. AGING FAILURE CAUSE RESULTS

In the failure-cause analysis, 2012 NPRDS records were meticulously examined to determine a failure cause for component failures. These component failures were associated with the auxiliary feedwater, Class 1E electrical power distribution, high pressure injection, and service water systems. The purpose of the failure-cause determination was to identify the aging mechanisms that caused the component failures. The depth of this analysis was limited to the level of resolution available in the NPRDS failure records. To determine a true root cause of failure in every case is beyond the scope of this study. Such a determination would require a detailed in-depth engineering evaluation to be performed on the components and the plant maintenance practices. Guidelines developed by the Root Causes of Component Failures Program[3] were used to evaluate the NPRDS failure reports and identify the failure mechanisms.

Failure cause is defined as the underlying discernable cause of failure contained in the failure report for a component. The failure-cause fractions, as used in this analysis, were derived for use in probabilistic risk assessment analyses and therefore, are component and failure mode specific. The cause fraction is simply the number of failures due to a given cause divided by the total number of failures due to all causes for the component and failure mode of interest.

The aging-related cause fraction data presented in this paper reflect an upper and lower bound derived from the operational data source. These bounds are developed to reflect the uncertainty encountered in accurate identification of aging-related causes from the component failure descriptions. An aging classification procedure was developed to aid the analyst in distinguishing aging-related from nonaging-related (random) failures. When insufficient information is contained in the failure description, the aging classification is unknown. These failures are then used to establish the upper and lower bounds for the aging-related cause fractions. The upper bounds contain the failures classified unknown as aging-related failures while the lower bounds contain them as nonaging-related failures.

The data for all the observed component failures for the systems studied are summarized in Table 3. The data are ordered by lower-bound aging fraction. The study determined (see Table 3) that the auxiliary feedwater system has a lower-bound fraction of 57% and an upper bound of 79% for aging-related failures, high

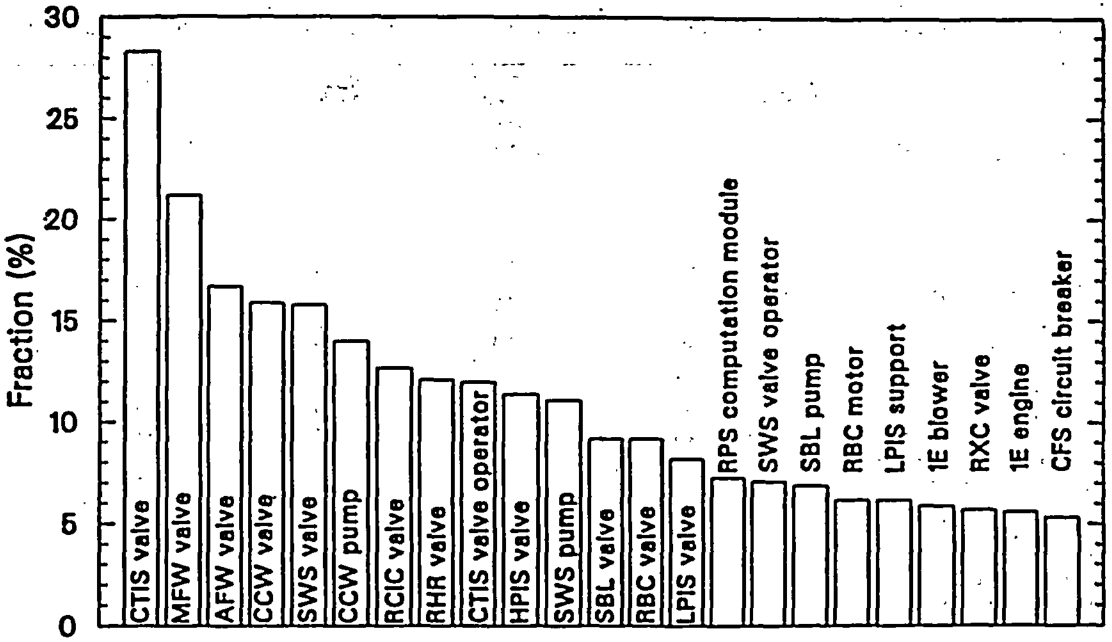


Figure 10. Ranking of components within systems studied in the aging survey by system level aging fractions.

pressure injection system exhibited 54% to 82%, and service water system exhibited 67% to 85%, respectively. The subsystems of the Class 1E electrical power distribution system exhibited the following upper- and lower-bound aging fractions: direct current (dc) power subsystem 28% to 57%, emergency onsite power subsystem 58% to 77%, and instrumentation and uninterruptible power supply subsystem 51% to 98%, respectively.

The failure data from this study for selected systems and components are summarized in Table 4. Only the dominant failure causes (counts >5) with their respective failure mode are itemized. The values in the total counts column refer to the total number of failures for the system-specific component and the specific failure mode listed. The upper- and lower-bound aging fractions are also indicated.

Examination of the total number of observed component failures in Table 3 indicates that the two auxiliary feedwater components having the highest potential impact on the system due to aging are pneumatic-operated valves and check valves. There were 100 failures examined for pneumatic-operated valves and 90 failures for check valves. The aging-related failure fraction for pneumatic-operated valves has a lower bound of 63% and an upper bound of 79%. The fraction for check valves has a range of 87% to 92%. Dominant failure causes for pneumatic-operated valves were wear and binding/out of adjustment which account for 39% and 16%, respectively, of the total pneumatic-operated valve failures. The dominant failure cause for check valves was wear, which accounts for 60% of the total check valve failures.

The dominant failure mode for pneumatic-operated valves was fails to close, which accounted for 49% of the pneumatic-operated valve failures. As indicated by the data in Table 4, the dominant aging-related failure causes for pneumatic-operated valve failures for the failure mode fails to close were wear (37%) and binding/out of adjustment (20%). The study determined that the subcomponents most affected by the wear mechanism were the valve operator and the valve internals. Similarly, the piece parts affected by the binding/out of adjustment mechanism were the limit and torque switches in the valve operator. Failures due to foreign materials intrusion accounted for 22% of the pneumatic-operated valve fails to close failures. The failure mode, fails to open accounted for 20% of the pneumatic-operated valve failures. The data presented in Table 4 indicate that the dominant aging-related failure cause for pneumatic-operated valve failures for the failure mode fails to open was wear (30%). The study determined that the subcomponents most affected by the wear mechanism were piece parts of the valve operator such as air regulators, limit switches, and couplings.

The dominant failure mode for auxiliary feedwater check valves was internal leakage, which accounted for 73% of the check valve failures. The failure mechanism, wear, accounted for 70% (see Table 4) of the check valve failures for the failure mode internal leakage. The study determined that the piece part most affected by the wear mechanism in check valves for the failure mode internal leakage was the valve seat.

The data presented in Table 3 also indicate that motor-driven and turbine-driven pumps in the auxiliary feedwater system have a relatively significant impact on the system performance due to total number of failures for these components (63 and 58, respectively). Aging accounted for 48% of the motor-driven pump failures and 47% of the turbine-driven pump failures. The dominant failure mode for both types of pumps was fails to run, which accounted for 52% of the motor-driven pump failures and 67% of the turbine-driven pump failures. The dominant aging-related failure cause for the fails to run failure mode for both types of pumps was wear. Wear accounted for 21% of the motor-driven pump fails to run failures and 28% of the turbine-driven pump fails to run failures. The study determined that, for motor-driven pump failures, the motor bearings were the dominant piece part that failed due to the wear mechanism. Similarly, the turbine was the subcomponent most affected by the wear mechanism in the turbine-driven pump failures, with the piece parts of the turbine being affected including seals, O-rings, packing, and governor valves.

TABLE 3. COMPONENT FAILURES AND AGING FRACTIONS AS DETERMINED BY THE REPORTED CAUSES OF FAILURE STUDY[a]

System/Components[b]	Total[c] Failures	Lower-Bound Aging Total	Upper-Bound Aging Total	Lower-Bound[d] Aging Fraction	Upper-Bound[d] Aging Fraction
Auxiliary Feedwater					
Relief valve	7	7	7	1.000	1.000
Check valve	90	78	83	0.867	0.922
Hand control valve	16	12	15	0.750	0.938
Snubber	7	5	5	0.714	0.714
Pneumatic-operated valve	100	63	79	0.630	0.790
Circuit breaker	8	5	7	0.625	0.875
Flow transmitter	27	14	22	0.519	0.815
Motor-operated valve	43	22	36	0.512	0.837
Flow control recorder	10	5	9	0.500	0.900
Motor-driven pump	63	30	42	0.476	0.667
Turbine-driven pump	58	27	38	0.466	0.655
Level controller	18	8	17	0.444	0.944
Pressure switch	21	9	18	0.429	0.857
Pressure controller	7	3	5	0.429	0.714
Flow controller	19	8	13	0.421	0.684
Relay	12	5	9	0.417	0.750
Level control indicator	20	7	14	0.350	0.700
Pressure transmitter	17	5	12	0.294	0.706
Support	5	0	4	0.000	0.800
Total	548	313	435		
System aging fractions				0.571	0.794
Class 1E Electrical Power Distribution					
DC Power Supply Subsystem					
Inverter	63	18	38	0.286	0.603
Circuit breaker	5	1	1	0.200	0.200
Total	68	19	39		
Subsystem aging fractions				0.279	0.574
Emergency Onsite Power Supply Subsystem					
Diesel generator	113	66	86	0.584	0.761
Circuit breaker	5	2	5	0.400	1.000
Total	118	68	91		
Subsystem aging fractions				0.576	0.771
Uninterruptible Power Supply Subsystem					
Battery charging unit	35	18	34	0.514	0.971
Battery	10	5	10	0.500	1.000
Total	45	23	44		
Subsystem aging fractions				0.511	0.978
High-Pressure Injection					
Relief valve	18	17	17	0.944	0.944
Snubber	10	9	10	0.900	1.000
Check valve	18	15	15	0.833	0.833
Pneumatic-operated valve	10	8	10	0.800	1.000
Hand control valve	12	7	9	0.583	0.750
Circuit breaker	11	6	9	0.545	0.818
Pressure transmitter	6	3	6	0.500	1.000

TABLE 3. (continued)

System/Components[b]	Total[c] Failures	Lower-Bound Aging Total	Upper-Bound Aging Total	Lower-Bound[d] Aging Fraction	Upper-Bound[d] Aging Fraction
Heat tracing heater	21	10	14	0.476	0.667
Motor-driven pump	17	8	12	0.471	0.706
Motor-operated valve	70	32	53	0.457	0.757
Flow transmitter	17	7	17	0.412	1.000
Load sequence controller	10	4	6	0.400	0.600
Level transmitter	21	4	20	0.190	0.952
Total	241	130	198		
System aging fractions				0.539	0.822
Service Water					
Flow switch	16	16	16	1.000	1.000
Check valve	31	27	28	0.871	0.903
Strainer	21	17	18	0.810	0.857
Motor-driven pump	167	129	140	0.772	0.838
Pneumatic-operated valve	47	36	45	0.766	0.957
Pressure indicator	6	4	6	0.667	1.000
Hand control valve	17	11	12	0.647	0.706
Flow indicator	5	3	4	0.600	0.800
Circuit breaker	17	9	12	0.529	0.706
Motor-operated valve	111	43	93	0.387	0.838
Total	438	295	374		
System aging fractions				0.674	0.854

---

a. Data have been listed by lower-bound aging fractions.
b. Data used exclude components with <5 total failure counts.
c. Data used exclude failure records for which failure cause was unclassifiable.
d. Fraction denominator is system-specific component failure total.

TABLE 4. REPORTED FAILURE-CAUSE IDENTIFICATION SUMMARY

System/Components	Failure Mode	Total Counts	Dominant Failure Cause	Failure Cause Fraction	Lower-Bound Aging Fraction	Upper-Bound Aging Fraction
Auxiliary Feedwater						
Check valves	External leakage	13	Wear	0.615	0.615	0.615
	Fails to operate as required	8	Design error	0.750	0.750	0.750
	Internal leakage	66	Wear	0.697	0.697	0.697
			Human maintenance error	0.106	0.000	0.015
Hand control valves	External leakage	6	Wear	1.000	1.000	1.000
Motor-driven pumps	External leakage	14	Wear	1.000	1.000	1.000
	Fails to run	33	Wear Water intrusion	0.212 0.212	0.212 0.000	0.212 0.000
Pneumatic-operated valves	External leakage	8	Wear	0.875	0.875	0.875
	Fails to close	49	Wear Binding/out of adjustment Foreign materials intrusion	0.367 0.204 0.224	0.367 0.082 0.020	0.367 0.184 0.041
	Fails to open	20	Wear	0.300	0.300	0.300

TABLE 4. (continued)

System/Components	Failure Mode	Total Counts	Dominant Failure Cause	Failure Cause Fraction	Lower-Bound Aging Fraction	Upper-Bound Aging Fraction
	Fails to operate as required	10	Wear	0.600	0.600	0.600
Turbine-driven pumps	Fails to run	39	Wear	0.282	0.282	0.282
			Binding/out of adjustment	0.179	0.026	0.179
Class 1E Electrical Power Distribution						
DC Power Supply Subsystem						
Battery chargers	Loss of function	35	Faulty module	0.657	0.400	0.657
Emergency Onsite Power Supply Subsystem						
Diesel generator	Fails to run	43	Wear	0.326	0.326	0.326
	No failure	49	Water intrusion	0.184	0.041	0.061
			Wear	0.122	0.122	0.122
			Cyclic fatigue	0.102	0.102	0.102
Instrument and Uninterruptible Power Supply Subsystem						
Inverters	Loss of function	63	Design error or inadequacy	0.079	0.000	0.016
			Wear	0.079	0.079	0.076
			Electrical overload	0.143	0.000	0.016
			Faulty module	0.286	0.079	0.222
			Short circuit	0.127	0.016	0.079
High-Pressure Injection						
Check valves	Internal leakage	15	Wear	0.600	0.600	0.600
Motor-operated valves	External leakage	7	Wear	0.857	0.857	0.857
	Fails to close	20	Binding/out of adjustment	0.300	0.100	0.250
Service Water						
Check valves	Internal leakage	25	Wear	0.480	0.480	0.480
			Corrosion	0.240	0.240	0.240
Motor-operated valves	External leakage	7	Wear	0.714	0.714	0.714
	Fails to open	27	Binding/out of adjustment	0.296	0.037	0.222
	Fails to close	43	Wear	0.140	0.140	0.140
			Binding/out of adjustment	0.535	0.000	0.488
	Fails to operate as required	17	Binding/out of adjustment	0.294	0.059	0.235
Pneumatic-operated valves	External leakage	5	Wear	1.000	1.000	1.000
	Fails to close	15	Wear	0.333	0.333	0.333
	Fails to operate as required	13	Wear	0.385	0.385	0.385
Motor-driven pumps	Fails to run	89	Wear	0.326	0.326	0.326
			Binding/out of adjustment	0.112	0.011	0.067
			Foreign materials intrusion	0.258	0.180	0.180
	External leakage	64	Wear	0.703	0.703	0.703
			Foreign materials intrusion	0.141	0.125	0.125
Strainers	Loss of function	13	Wear	0.615	0.615	0.615
	Plugged	8	Foreign materials intrusion	0.750	0.500	0.500

Examination of the total number of observed component failures in Table 3 indicates that the component in the high pressure injection system with the highest potential impact on system performance is the motor-operated valve. There were 70 failures reported for motor-operated valves. The dominant failure cause for motor-operated valves was wear. Wear accounted for 28.5% of the total motor-operated valve failures in the high pressure injection system followed by binding/out of adjustment (20%), human maintenance error (13%), and foreign materials intrusion (11.5%). The study determined that the subcomponent most affected by the wear mechanism was the valve operator. Furthermore, binding/out of adjustment affected the valve operator's torque and limit switches and foreign materials intrusion failed the valve operator's contacts.

The dominant failure modes for high pressure injection motor-operated valves were fails to open/fails to close (34%) and fails to close (29%). None of the failure causes for the failure mode fails to open/fails to close were determined to be significant because their failure counts were not greater than 5. For the failure mode fails to close, binding/out of adjustment was the dominant failure cause accounting for 30% of these failures. These closure failures were dominated by failure of the valve operator. The piece parts, most affected were the torque and limit switches.

Examination of the total number of observed component failures in Table 3 indicates that the service water component having the highest potential impact on the system due to aging is the motor-driven pump. There were 167 failures reported for motor-driven pumps. The dominant failure causes for motor-driven pumps are wear (45%) and foreign materials intrusion (21%). The subcomponents most affected by the wear mechanism were the pump seals and packing. The foreign material intrusion failures were primarily due to sand and other particulates sometimes found in influent service water.

The dominant failure modes for service water motor-driven pumps are fails to run (53%) and external leakage (38%). Wear and foreign materials intrusion are the dominant reported failure causes of these two failure modes. The data presented in Table 4 indicate that wear accounted for 32.6% of the fails to run failures and 70.3% of the external leakage failures; and foreign materials intrusion accounted for 14% of the fails to run failures and 25.8% of the external leakage failures. Inspection of the failure records pertinent to service water system motor-driven pumps indicated that the failures for the failure mode external leakage were caused by wear of the pump seals and packing or foreign materials intrusion into the pump seals. The study determined that for the failure to run failures of motor-driven pumps, the wear mechanism was associated with the pump internals such as bearings, impeller, and shaft. Many of these failure reports stated internal wear or general pump wearout as the cause of failure. The wear mechanism was also associated with the motor of the motor-driven pump and usually affected the motor bearings. Sand and other particulates found in the influent water was the dominant cause of foreign material intrusion failures.

The service water components with the second highest potential for aging impacts on system operation are motor-operated valves. There were 111 failures reported for motor-operated valves (see Table 3). The resulting potential aging-fraction range is 39% to 84%. The largest number of failures occurred for the failure mode fails to close. Inspection of the failure records indicated that the dominant aging failure-cause wear often caused the failure of the valve stem connection to the valve operator. The data presented in Table 4 further indicate that the failure cause out of adjustment for motor-operated valves displays a potentially high aging contribution for the failure modes fails to close, fails to open, and fails to operate as required. Examination of the failure records reveals that out of adjustment was associated with the valve operator and the time requirement for actuation of the valve movement.

Service water check valves are significantly affected by aging. The largest potential aging fraction was calculated for service water system check valves (Table 3, 87% to 90%). The dominant failure mode is internal leakage (81%), with dominant aging failure causes being wear (48%) and corrosion (24%). Examination of the failure records indicated that the piece parts most affected by wear were the valve seat and valve disc, as would be expected. Similarly, the piece parts most affected by corrosion were the valve body and the valve internals.

Examination of the total number of observed component failures in Table 3 indicates that the diesel generators are the largest single contributor of failures for the Class 1E subsystems. That is, in part, because of the component boundaries developed for the diesel generator for the failure-cause study. The component boundary for the diesel generator included all components associated with the diesel engine, generator, and required subsystems. These subsystems include the diesel lube oil, diesel fuel oil, diesel starting air, diesel cooling water, and engine exhaust. Aging-related failures constitute 58% to 76% of the total diesel generator failures within the emergency onsite power supply subsystem. The largest single aging failure cause for diesel generators is wear (19.5%), which could be induced by vibration.

The dominant failure mode for diesel generators is fails to run (38%). The dominant failure cause for diesel generator fails to run failures is wear (32.6%). The failure records indicate that the wear failures are distributed over four diesel generator subsystems: diesel cooling water, diesel fuel oil, diesel lube oil, and diesel starting air. The components most affected are valves and pumps.

The dominant component fractions (greater than 5% at the system level) with their associated upper- and lower-bound aging fractions are illustrated in Figure 11. Inspection of Figure 11 indicates that the components of importance in the auxiliary feedwater system are pneumatic-operated valves, check valves, motor-driven pumps, and turbine-driven pumps. Of these components, check valves have the highest aging-fraction range of 14% to 15% (calculated at the system level). The components in the service water system with the highest system importance are motor-driven pumps, motor-operated valves, and pneumatic-operated valves. At the system level, service water motor-driven pumps have an aging fraction range of 30% to 32%; service water motor-operated valves have a range of 10% to 21%; and service water pneumatic-operated valves have a range of 8% to 10%. Data for the Class 1E electrical power distribution subsystems indicate that battery charging units (40% to 76%) dominate the dc power supply subsystem; diesel generators (56% to 73%) dominate the emergency power supply subsystem; and inverters (26% to 56%) dominate the instrument and uninterruptible power supply subsystem. (The fractions for the

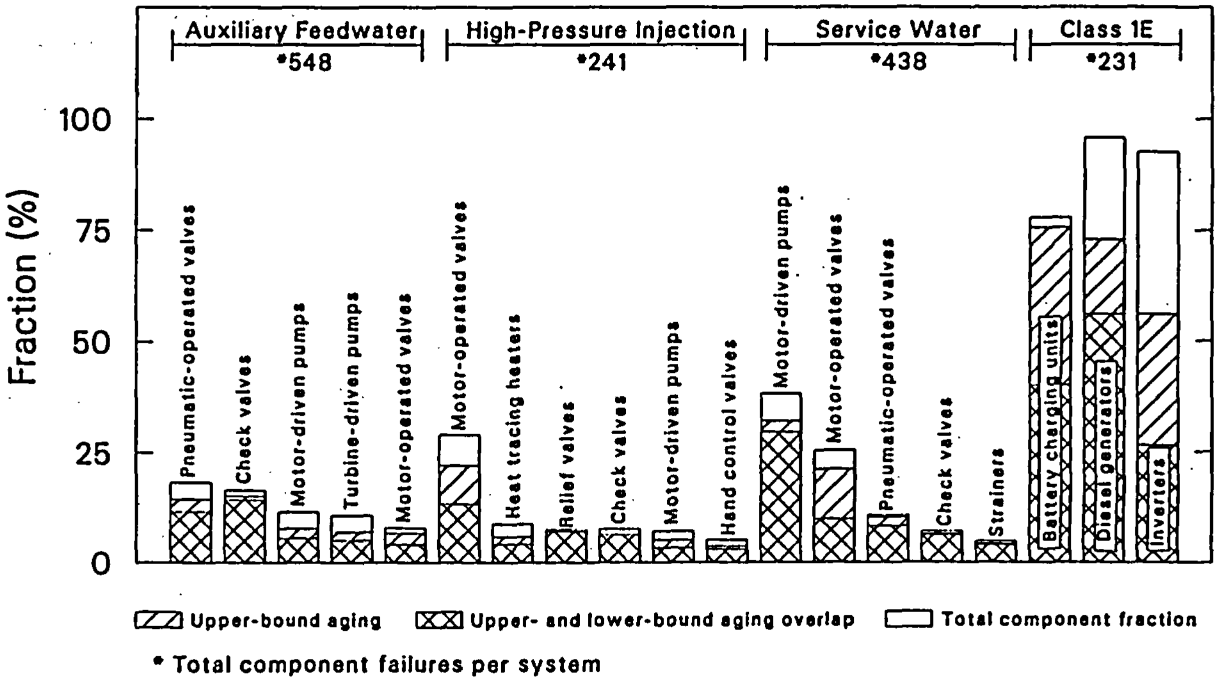


Figure 11. System fractions for dominant components with the systems studied in the reported failure-cause study.

Class 1E electrical power distribution subsystems are calculated using the total failures per subsystem as the denominator. Therefore, the sum of the fractions illustrated in Figure 11 for the Class 1E will be greater than one.)

The dominant failure cause fractions (greater than 5% at the system level) with their associated upper- and lower-bound aging fractions are illustrated in Figures 12 and 13. The data presented in Figure 12 indicate that the failure cause wear dominated all the fluid systems. Wear accounted for 30% of the failures in the auxiliary feedwater system, 21% in the high pressure injection system, and 32% in the service water system. Data for the Class 1E electrical power distribution subsystems (Figure 13) indicate that faulty module dominated the failures in the dc power subsystem and the instrumentation and uninterruptible power subsystem, and wear dominated the failures in the emergency onsite power supply subsystem.

4. CONCLUSIONS

In the aging survey, aging fractions have been tabulated for components in fifteen PWR and BWR systems. Systems and specific components within the systems have been identified, using survey techniques, for which aging-related failures contribute significantly to the component failure rate. The results indicate that component aging contributed to approximately one-third of all the failures reported. The data also indicate that U.S. reactors are experiencing more aging-related failures in their normally operating fluid systems than in the standby systems. The analysis indicates that valves and pumps were the components most affected by aging-related failures.

In the reported failure-cause analysis, component failures in selected systems were analyzed to identify the mechanism or cause of the failure. The results indicate that the three fluid systems were significantly impacted by aging-related failures. Valves and pumps were the components in the fluid systems most affected by aging mechanisms. Failures of the diesel generator dominated the Class 1E electrical distribution system. The study indicates that wear was the dominate aging mechanism at both the component and system levels.

5. REFERENCES

1. Meale, B. M. Satterwhite, D. G. An Aging Failure Survey of Light Water Reactor Systems and Components, U.S. Nuclear Regulatory Commission Rep. NUREG/CR-4747, Vol. 2, EGG-2473, July 1988.

2. NPRDS Reporting Procedure Manual, INPO 84-011, April 1984.

3. Satterwhite, D. G. Cadwallader, L. C. Vesely, W. E. Meale, B. M. Root Causes of Component Failures Program: Methods and Applications, NUREG/CR-4616, EGG-2455, December 1986.

NOTICE

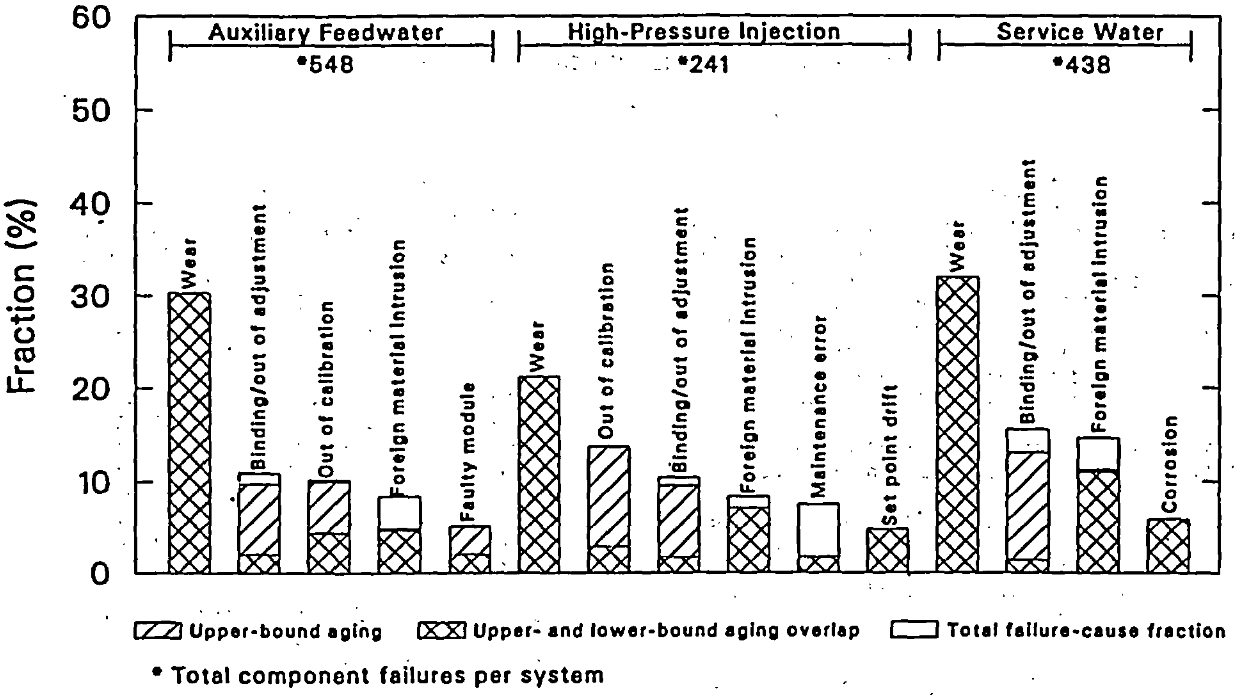


Figure 12. System fractions for dominant failure causes for auxiliary feedwater, high-pressure injection, and service water systems.

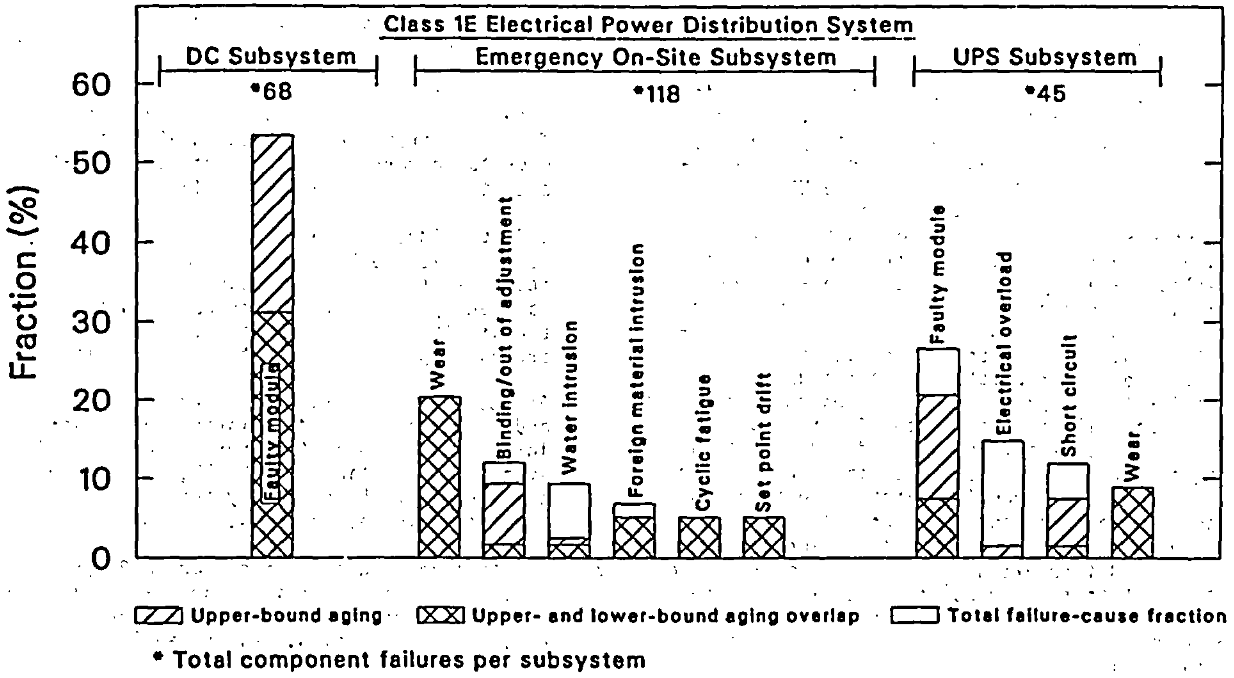


Figure 13. System fractions for dominant failure causes for Class 1E electrical power distribution system.

AGING AND NRC DATA SOURCES

Victor Benaroya, NRC

Introduction

The Office for Analysis and Evaluation of Operational Data (AEOD) was formed in 1980 to provide independent review of operating experience in response to TMI-2 lessons learned. AEOD's focus and role in the regulatory process is to independently analyze operating experience and feed back the lessons learned to the NRC, the industry and the public. To achieve this objective, AEOD depends on several data sources.

To track and evaluate aging impacts on component reliability, the following information is needed:

1. Component Engineering Data. Plant(s) where the component is installed, and equipment specification such as the system and model or other characteristics so that any additional information can be obtained;

2. History of the Component. Information needed includes the date of installation of the component, the dates of any failures, the cause of the failure, the date(s) if the component has been replaced. Data on corrective and preventive maintenance, and the surveillance tests performed on the component to track the history of that item, to help determine the effects in controlling the impact of aging on component reliability; and

3. Environment. The actual operating environment the component experienced, to understand whether additional aging occurred due to radiation, temperature, humidity, etc.

Data Sources

In support of the NRC aging and plant life extension program, and to track and evaluate aging impacts on component reliability, the following data sources are available to the NRC:

1. Nuclear Plant Reliability Data System (NPRDS) is a large computerized database that contains component engineering reports (approximately 500,000) and component failure reports (over 65,000) maintained by INPO. INPO members and the NRC have access to it. Each component engineering report contains descriptive information such as manufacturer, model number, the systems where it is installed, capacity, rating, etc. In addition, it may also contain information on testing, such as the method and frequency of test. When a component cannot fulfill its design function, a failure report is submitted. The failure report identifies the component and provides the date and time of failure, the length of time required to restore component function, and a description of the failure and corrective action. Reports are not submitted on preventive maintenance or maintenance due to incipient failures. Actual demand data are also missing. Some information needed for aging and life extension programs for specific plants, such as the impact of preventive maintenance, and the tracking of individual components throughout their service lives, generally cannot be obtained from the present NPRDS data.

   Participation in NPRDS by the industry is voluntary and, although most utilities participate, the quality of reporting varies considerably. In many cases, the cause description codes, if identified, do not reflect the mechanism causing the failure, but are instead related to the effect of the failure. Industry, with INPO's urging, is improving the reporting record.

   Recently, EPRI has provided specific funds to support the NPRDS rescoping efforts to accommodate additional data needed for reliability studies and license renewal. Maintenance performance indicators may also require keeping better records. When these initiatives are implemented, we expect NPRDS to improve significantly.

   At present, NPRDS is not adequate in itself to supply all data necessary for plant specific aging evaluations.

2. The Sequence Coding and Search System Review (SCSS) database contains LERs reported by utilities since 1980. This system provides a structured format for detailed coding of component, system, and unit effects, as well as personnel errors.

   Each individual occurrence reported in the LER, e.g., component failure, system failure, personnel action, etc., is coded as a separate step in the overall sequence using information from the entire LER; i.e., the narrative, abstract, and coded fields. Specific information, such as cause, system involved, component involved, component vendor, effect, etc., for each step is encoded. These individual steps are linked together in chronological order to construct an overall event sequence. The relationships between steps, i.e., predecessor and successor relationships, can be searched as well as the detailed information of individual steps. Consequently, one can query SCSS for such things as feedwater transients preceding RPS actuations, operator errors occurring subsequent to RPS actuations, and other sequential relationships.

   SCSS provides the ability to do general or broad searches at the LER level and more specific searches at the step or sequence level. However, many equipment failures are not reported in LERs, because items such as single equipment failures are not reportable under the LER Rule (50.73).

   An SCSS search could be complementary to NPRDS and might help provide information and insight into some equipment failures. However, SCSS, even in conjunction with NPRDS, is not adequate to supply all the necessary information for plant specific aging evaluations.

3. The Accident Sequence Precursor (ASP) program reviews LERs and identifies and ranks precursors to severe core damage (inadequate core cooling) accidents. Operational events evaluated are limited to:

- Events that involved the failure of at least one system required to mitigate a core damage initiator (i.e., a loss of feedwater, loss of offsite power, small-break LOCA, or steam-line break);
- Events that involved the degradation of more than one system required to mitigate one of the above initiating events; or
- Events that involved an actual initiating event which required safety system response.

Events typically not addressed due to low significance and programmatic constraints include uncomplicated reactor trips, losses of feedwater without additional failures, and single failures in mitigating systems. With the exception of initiating events, precursors typically involve events not bounded by the single failure design basis applied to safety-related systems in the United States.

Observed failures initiating event frequency and system failure probability estimates are used, in conjunction with event trees depicting potential paths to severe core damage, to estimate a conditional probability associated with each precursor. This probability is an estimate of the chance of subsequent severe core damage, given that the failures observed during the event occurred in the manner they did, and can be considered a measure of the residual protection available during the event. The conditional probabilities associated with each precursor are used to rank precursors according to significance. The more serious events are useful in the identification of both dominant sequences to severe core damage and unusual events not typically addressed in probabilistic risk assessment.

To distinguish differences among system designs, train based system models are used in conjunction with plant-class event sequences to more accurately represent each plant. These system models include the potential for restoration of initial failures and required operator initiation. The difference between failures and unavailabilities, from a common-mode standpoint, is addressed in the treatment of train conditional probabilities.

The current plant-class specific event sequence and train-specific system models permit operational data to be apportioned among relevant plants while still permitting a reasonable estimate of the core damage significance to be calculated with consideration of the plant design.

The ASP program identifies and evaluates the conditional core melt probability of significant events at operating plants. This information can be used in the aging program to focus attention on issues that have shown a large impact on plant safety, either categorized for individual plants or on a generic basis. Remedial work on the areas of exhibited importance to safety may restore some of the safety margin lost by the aging process; but does not replace consideration of those issues that are of more direct concern to aging and not reflected in the database because of longer wearout times.

ASP results could also be used quantitatively to assess the "goodness" of the prior operation of the plant. Some plants may be judged not to be good candidates for continued operation because of their poor past performance. To develop this into licensing criteria would need considerable effort since probabilistic perspectives have not been embraced unconditionally in the past as bases for regulatory decisions.

The event analysis program (ASP - the computer program) does not lend itself to examining aging issues because it is designed to process operating events. There are other computer codes such as SARA and IRRAS which may be helpful in examining the impact of plant component failure data and/or new issues. The component failure data could be simply used to estimate core melt likelihood using the stylized sequences in SARA. More again, there is a probabilistic measure of a form of prior plant performance (does not include plant specific human errors which may be dominant and unrecoverable). New issues could be examined with the IRRAS code. However, both of these codes suffer from limited plant models, and therefore may not be usable for all plants or plant types.

4. Foreign reactor event reports are processed and entered on a database that provides an abstract and key word search capability. This database could be searched to identify topics of potential interest to the aging program. Hard copies of specific reports are available from International Programs. These reports are proprietary so that they cannot be cited directly. However, the database is still useful for capturing issues of interest.

AEOD Studies

AEOD has several studies, i.e., Feedwater Regulating and Bypass Valves, Feedwater Pumps, that while studying trends and patterns, the effects of aging were a factor. Findings of these studies could be considered during license renewal requests.

Performance Indicators

The existing performance indicator program may provide information that is beneficial to aging and life extension issues. The NRC performance indicator (PI) program provides an objective view of operational performance and enhances our ability to recognize changes in the safety performance of operating plants earlier.

Some of the data used by the PI program may be of particular relevance to aging and life extension program needs. In particular, safety system failure (SSF) and equipment forced outage (EFO) data may be useful for monitoring failures and/or degradations of systems that are the result of aging or "component wearout." Together with the cause code information, these data may help identify areas that require additional scrutiny. For example, a unit may have a history of diesel generator failures. In this case, the life extension application review effort should devote a proportionate amount of resources to the adequacy of the diesel generators. Used in this manner, the data would serve as a tool to help prioritize review efforts.

AEOD is in the process of developing and incorporating additional performance indicators, including a maintenance indicator. Since the maintenance indicator has not been fully defined at this time, it is difficult to predict how beneficial the data for this indicator may be for providing information pertinent to aging and life extension. In any case, it is quite possible that the data may correlate well with component or system mean life. In this case, the data would be useful for predicting the mean time to failure of major safety systems. Safety system unavailabilities could be determined from train level or component level data, whether it is obtained from an upgraded NPRDS or as the result of a 10 CFR 50.74 rule now under consideration. For example, if the safety system mean time to failure (at previous maintenance levels) is less than the life extension period, then safety system upgrade or an enhanced maintenance program might be required as a license condition for life extension.

In summary, data from the performance indicator program can provide useful information relative to aging and life extension. The information could be used, along with other inputs, in the decision making process for license extension. Tracking of particular data, such as that used for SSF and EFO performance indicators, may help reviewers prioritize review efforts. Data for other performance indicators currently under development, particularly a maintenance indicator, would most likely provide additional pertinent information relative to aging and life extension. It is suggested that the performance indicator and aging and life extension programs be reviewed on an ongoing basis as the programs evolve, consistent with Commission guidance on the use of performance indicators.

Conclusions and Recommendations

In summary, the present database needs improvements for use as a basis to make knowledgeable decisions for life extension. One way to achieve a more complete database is by better participation in the NPRDS data bank.

USING DATA BASES TO UNDERSTAND AGING*

M. Subudhi and R. Lofaro

Abstract

A great deal of nuclear power plant operating experience has been documented in the form of national data bases. Individual plant maintenance records, design and manufacturing data, equipment qualification documents, and other published reports provide an enormous amount of information that can characterize a component. This can be of great importance in examining and understanding the effects of aging on plant performance and safety. To obtain meaningful results, however, the strengths and weaknesses of these data sources must be recognized and accounted for.

This paper discusses the various data bases currently available. The type of information provided in each is explained, as well as the advantages and limitations of each data source. Their application to aging analyses are discussed. The need for a review process of a data source to utilize information for aging analyses is also included. Validation of the data using plant specific data is also discussed.

Aging characteristics of components, systems or structures in nuclear power plants are discussed. Examples are presented to demonstrate the feasibility of obtaining this information from the various data sources available. Results from the aging studies performed at Brookhaven National Laboratory under the Nuclear Plant Aging Research (NPAR) are used for illustrations.

Introduction

Aging is a degradation process (or mechanism) which exists at every level in a plant's hierarchy. If unchecked, it can limit the life of a component, system, or structure, and increase the risk to plant safety. Therefore, understanding aging phenomena is important for the safe operation of a nuclear plant.

Degradation mechanisms attributed to aging occur in materials subjected to certain stress conditions over a period of time. These processes are well understood when one type of material is exposed to one kind of stress condition. However, with the complexities of composite materials, or a component made of many different materials (which is the actual case for most component designs), and the synergistic effects of several stress conditions, these processes are difficult to understand. Extensive laboratory testing and material analyses[1] are necessary to characterize these complex phenomena. Also, since aging is a time-dependent process, considerable time would be necessary to completely understand the true characteristics in a plant environment and operating conditions. Since this approach is not feasible, an alternate approach is to analyze the existing 15-20 years of plant experience to understand the trends and update it as the data become available. Although extrapolation of these trends for future years may not provide the actual performance, an intelligent analysis of the first few years of history could indicate the direction the plant might go without taking any additional corrective actions.

Under the auspices of the Nuclear Regulatory Commission (NRC), Brookhaven National Laboratory (BNL) has been conducting aging research of components and systems. This work is performed under the Nuclear Plant Aging Research (NPAR) program and is a multiphase approach to understand aging behavior and recommend ways of mitigating the effects. BNL has completed studies on electric motors, battery chargers and inverters, and reactor coolant pump seals. The phase 1 studies on circuit breakers and relays, and motor control centers (MCCs) are also completed. System studies include the phase 1 study of component cooling water (CCW) systems in PWR, and on-going programs on residual heat removal (RHR) systems in BWR and instrument air system. In addition, several other special studies are in place to characterize the aging of components and how to use them in NRC inspection audits. BNL has successfully utilized the available data sources to perform this research and this paper is intended to present the strengths and weaknesses of these data bases and how to account for them in aging analyses.

Data Sources

Four different kinds of data sources are utilized to understand aging effects in components and systems; 1) design and manufacturing data for determining the materials used, 2) plant design data for environmental and operating parameters, 3) plant maintenance and surveillance activities, and 4) plant operating experience data to develop the aging trends.

To utilize effectively the last data source, one has to understand elements from the first three data sources, since the data provide the actual operating experience trends of components in nuclear power plants. The NPAR program plan[2] has cited the following data bases for obtaining the operating data:

- Licensee Event Reports (LERs)
- In-Plant Reliability Data Systems (IPRDS)
- Nuclear Power Experience (NPE)
- Nuclear Plant Reliability Data Systems (NPRDS)

Other researchers[3,4,5] have been using different data sources such as plant specific data, incident investigation reports (IIRs), NRC information notices and bulletins, plant maintenance records, and industry based research reports from both domestic and foreign utilities. All of these provide information pertinent to understanding the aging characteristics of components.

The approach adopted at BNL for the NPAR program (Figure 1) utilizes all the data sources mentioned above. To understand the aging degradation in any component or system, the three key elements one has to determine are 1) the age-sensitive materials present, 2) the type of conditions these materials are subjected to, and 3) if there exist any catalytic factors which can accelerate or decelerate the degradation process.

---

* This work done under the auspices of the U.S. Nuclear Regulatory Commission.

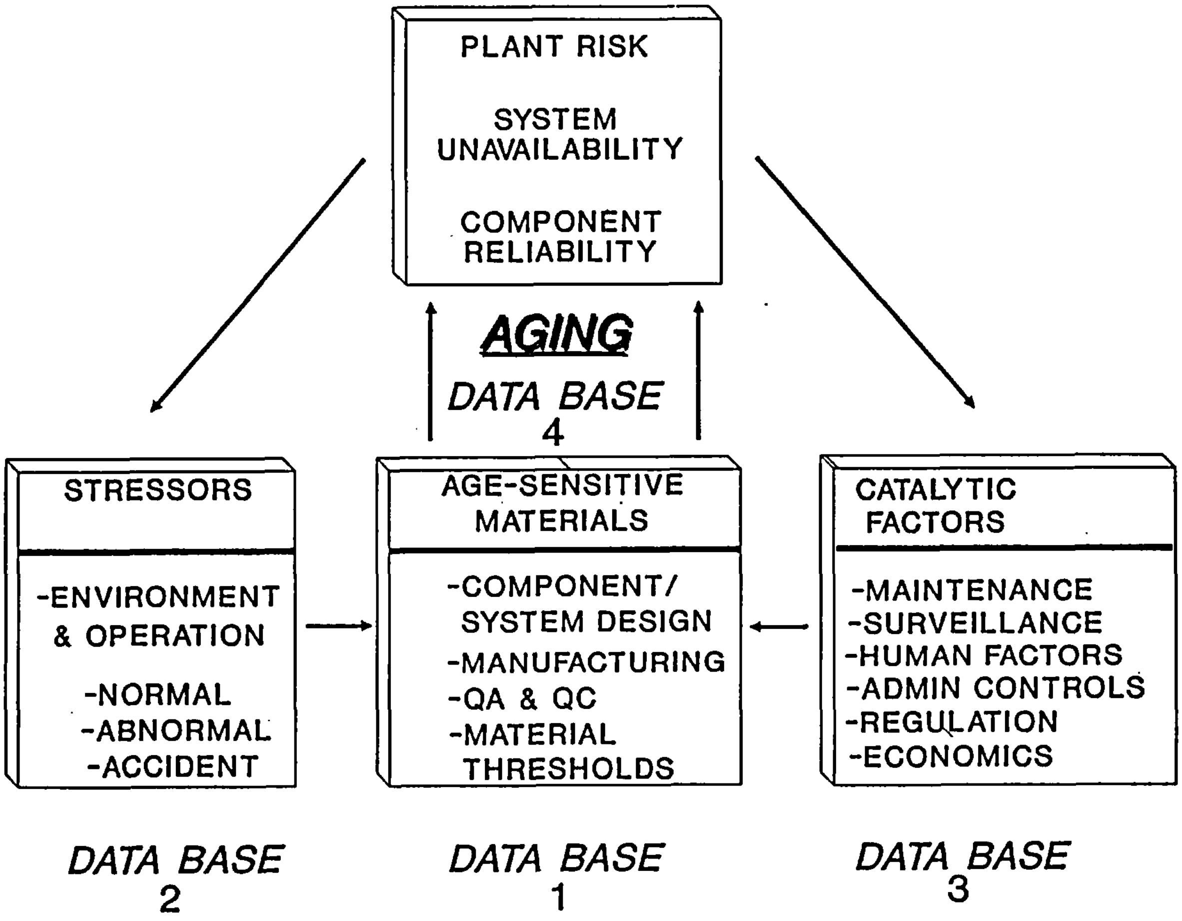


Figure 1 Elements for Understanding Aging

The material data are typically available from the manufacturers or qualification documents for components. Final Safety Analysis Reports (FSARs) are used to obtain the system design and the components included in the system. The stress conditions are determined from the plant design parameters, typically from FSARs or equipment qualification (EQ) documents. Both environmental and operational parameters are reviewed for normal, abnormal and accident conditions of the plant operation. In general, these two elements are the basis for age-degradation of components and systems, and have very little room for enhancement without going through major design modifications of the plant.

The last element, which acts as a catalytic agent in increasing or decreasing the aging process, includes the actual maintenance and surveillance programs already in effect in the plant and other statutory or administrative commitments. This is the only element one can alter relatively easily with the age of the plant, if the component functional indicators or the system performance indicators indicate any sudden departure from normal trends. By tuning this and monitoring the results via trending, one can mitigate the aging effects.

Once these three elements are properly understood, the plant operating experience data available in various national data bases, mentioned earlier, are used to determine the following:

- Is aging a factor?
- Components or subcomponents that fail more frequently due to aging.
- Materials responsible for degradation.
- Failure modes, causes, and mechanisms.
- Effect on plant safety and operation.
- Are the catalytic factors effective?
- Aging trends of components, subcomponents or systems.
- How to improve and mitigate aging effects.

The data bases have evolved with time, changing in response to user needs. Comparison may be complicated by the diversity among the sources; therefore, an integration of appropriate data from each source provides the best results, specifically the operating experience data. The design data involving age-sensitive materials and stressors and the plant maintenance information are less diverse. The manufacture of a typical component usually includes the same types of materials, and plant designs provide similar stressors. The maintenance of a component is typically not very different from plant to plant.

The operating experience data bases provide a broad base to assess trends for the overall population. Frequently the LERs or NPRDS reports from a single plant provide too small a sample to reach any significant conclusions, making the use of aggregate experience desirable. Among the four data bases mentioned earlier, LER and NPRDS are the two most frequently used for aging analyses. The NPE data base is similar to that of LER, while the IPRDS data base is very limited and incomplete.

Figure 2 shows an example of the differences in failures reported to the LER and NPRDS data bases for the MCC. This is partly because of the fact that LERs relate to safety systems alone and partly due to the changes in LER reporting requirements in 1984 which limited the scope of reportable events. From this example it is evident that not all failures are reported to all data bases. This is one data base limitation that must be accounted for when analyzing the number of failures. One method of accounting for unreported failures is to compare actual plant records from one or more plants to the data base in question. From this comparison, the percentage of failures reported to the data base can be determined and used as a correction factor.

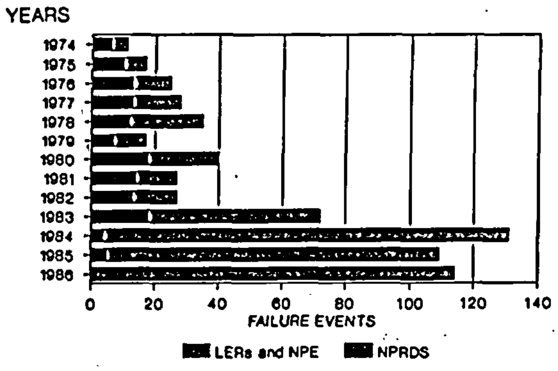


Figure 2 MCC Events Reported in LERs and NPRDS

Another limitation of the data bases is the inconsistencies between data bases, and also between different reporters to the same data base, in interpretation of codes, definitions of terms, and general understanding of the events leading to failure. Experience with the various data bases has shown that reports can sometimes include an incorrect identification of the failed component or description of events leading to the failure. This can be attributed to several reasons including a lack of standardized definitions, terminology and reportability for the data bases, as well as differences in experience and knowledge between personnel filing the reports. The method used by BNL to mitigate the effects of these inconsistencies is to have all data undergo a review process. The review is performed by a team of engineers using consistent definitions and terms. Although labor intensive, the review process allows more meaningful and accurate results to be obtained from the data.

Definition of Aging

In general, according to IEEE Std-100, aging (mechanism) is defined as any process attributable to service conditions which results in degradation of an equipment's ability to perform its functions. To understand this aging one needs to know the age-sensitive materials associated with the design of the component and the types of stress conditions these materials are subjected to. All data bases so far studied use this definition in categorizing the failure as "age-related." The nuclear industry also recognizes this aging definition in characterizing the process.

The NPAR program definition of aging, however, covers a broad view of the degradation process. In addition to recognizing the above definition of aging, it also includes the catalytic factors illustrated in Figure 1 as contributing factors for aging degradation. Under NPAR the term "aging" represents the cumulative changes with passage of time that may occur within a component, system or structure. This degradation takes place because of the following factors:

- natural internal chemical or physical processes during operation,
- external stressors caused by the storage or operating environment,
- service wear,
- excessive testing, and
- improper installation, application or maintenance.

In order to properly categorize and analyze the aging of a component or system, one has to consider the external factors which could contribute to the process by accelerating or decelerating the degradation. This requires a thorough review of all data available in any operating experience data base and sorting into appropriate categories.

Examining the information in any of the data bases, it is sometimes very difficult to obtain meaningful results using the above definition of aging. The aging fractions of component or system failures are always higher when the NPAR definition is used. This is obvious from the fact that many age-related failures can be attributed to improper maintenance, lack of proper procedures, tech spec violations, and other human factor problems.

Aging Assessment

To perform aging assessments, design data are obtained from sources such as vendor reports, tours to a manufacturing plant or nuclear power plant, FSARs, equipment qualification documents, and other pertinent published information. These provide the materials susceptible to aging and the stressors under normal, abnormal, and accident conditions. An assessment of the sensitivity to deterioration with age is conducted based on this information. Table 1 shows an example of such an analysis applicable to MCCs.[6] Plant maintenance manuals and interviews with plant or manufacturing personnel provide information on the expected rate at which various subcomponents or components degrade with age under various operating and environmental conditions.

Table 1 Typical Materials of Construction for MCC Subcomponents

Subcomponent	Typical Materials	Sensitivity Degradation With Age
Molded Case Circuit	Phenolic	Low
Breakers	Vulcanized Fiber	Low
Relays	Neoprene	Medium
Starter/	Lubricants	High
Contactor	Adhesives	High
	Steel	Low
	Silver Alloy Plating	Low
	Copper	Low
	Silicone	Medium
	Polyester	Medium
Transformer	Phenolic	Low
Coils	Varnish	Medium
	Fiberglass	Low
	Polyester Film	High
	Polyamide-imide Insul.	High
	Copper Wire	Low
	Teflon	Low*
Terminal	Phenolic	Low
Block	Phenolic	Low
Overloads	Silver Plating	Low
	Copper	Low
	Lubricants	High
	Vulcanized Fiber	Low
Cabinet	Steel	Low

* Radiation environments could increase sensitivity to medium.

Once this knowledge is acquired, the actual operating experience data is analyzed to understand the aging phenomena. Different sorts are conducted to identify the components that most often fail. Figure 3 illustrates one sort for components in the component cooling water systems in PWR plants.[7] Valves, pumps and heat exchangers dominate the system failures, which is typical for a continuously operating system with mechanical components. As evident from Figure 4, loss of function, tech spec violation, and leakage are the predominant modes of failure. Figure 5 shows wear, calibration, drift, and contamination are the dominant aging mechanisms contributing to these failures. Based on these sorts, all failure modes, causes, and mechanisms are assimilated for further analysis.

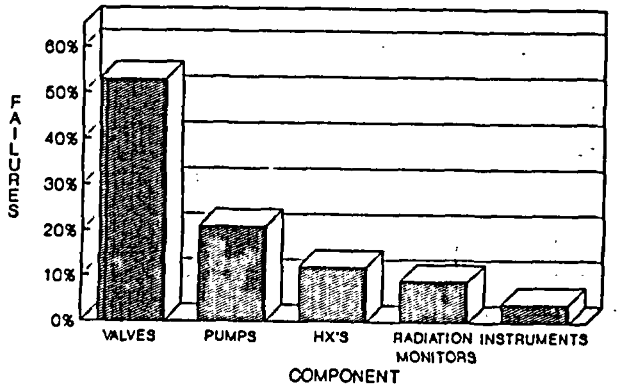


Figure 3 - CCW Component Failures - LERs

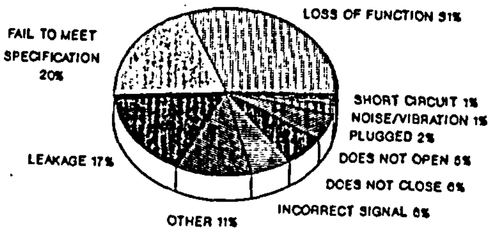


Figure 4 - CCW System Failure Modes - LERs Data

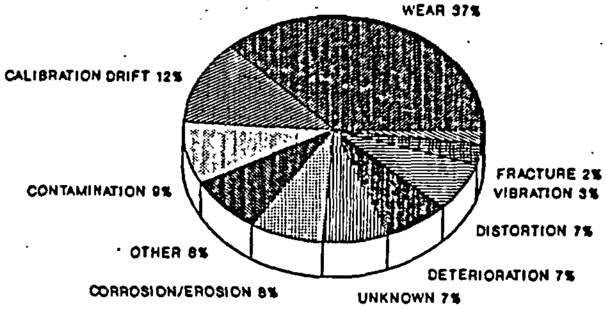


Figure 5 - CCW System Failure Mechanisms - NPRDS Data

The data bases also provide information on the effects of these failures on the component or system reliability. These include other systems affected as a result of a component or system failure. Some failures may lead to a reactor trip which has the potential for impacting safety because of the additional equipment and operator actions generally needed to bring the plant to a safe and controlled condition. An analysis of LER data for battery chargers and inverters for the period of 1984--1986 revealed that 16% of the reactor trips had post-scram recovery complicated by equipment failure or personnel errors unrelated to the cause of the trip.[8] The increase in the number of reactor trips due solely to inverter failures was applied to a core melt frequency calculation for several plants indicating an increase in core melt risk frequency at most of the plants selected. Also, the data indicated that the causes of reactor trips are often linked to feedwater and turbine-generator control systems. Thus, analysis of the operating experience helps understand the important factors contributing to component failures.

An additional application of the data bases is development of aging failure frequency or failure rate curves with age of the component. This is difficult to achieve since the history of each failure is not known. Assuming that pumps in all plants are subjected to similar operating and environmental conditions, and all repairs and maintenance performed in any pump is intended to bring it to a "good-as-old" condition, one can develop aging curves, as shown in Figure 6 for the CCW pumps. Additional analysis on uncertainty bounds and other factors can be mathematically developed for further use of this in PRA or reliability calculations. A system level unavailability curve can then be developed once the failure rates on various components within a system are known. This is demonstrated in Figure 7 for the CCW system. The prediction beyond 20 years is based on the extrapolation of each individual data assuming that no additional action will be taken to check the component failure rate due to aging.

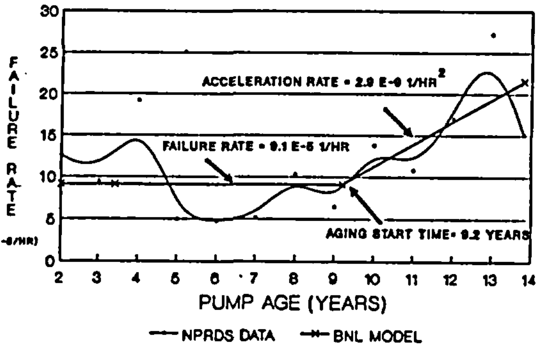


Figure 6 BNL Model For Pump Failure Rate

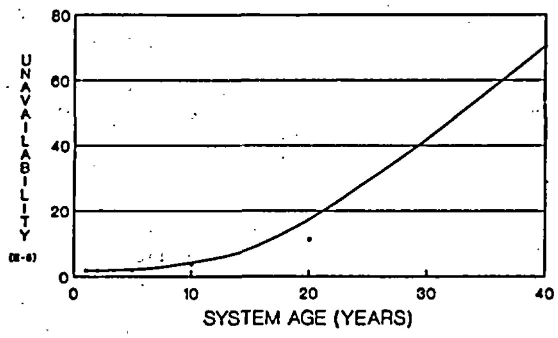


Figure 7 CCW System Unavailability Vs. Time

Conclusions

The national data bases have several virtues that make them suitable as sources of failure information. They contain a large amount of data representing a broad cross-section of nuclear power plants. The data is accessible, although sometimes difficult to obtain. Much of the data include sufficient information to identify basic failure characteristics, such as the component failed and the reason for failure. With proper review and evaluation, the data can also be used to identify prevailing trends.

Although a great deal of useful information is available from the data bases, there are limitations and weaknesses to it which must be recognized. In general, the data bases do not contain a complete record of all failures. This is partly due to the nature of the data bases and the failures required to be reported. The result is that failure frequencies determined directly from the data base information will probably be lower than actual. However, it must be noted that a large cross-section of plants is represented in the data bases. Using the data for analyzing failure characteristics, such as causes, modes and mechanisms, should not be severely affected by this deficiency. Using the data bases for evaluating aging effects is, therefore, a valid use of the data.

An additional concern with the data base information is the inconsistency of 1) the interpretation of codes used to report events, and 2) the understanding of the events associated with the failure. However, its effect on analysis results can be mitigated by 1) performing a thorough review of the data, and 2) validating the results by comparison with actual plant data. By performing an independent review using consistent definitions and interpretations, the data base information can provide meaningful results. The results should then be compared with findings from actual plant data to ensure that erroneous trends or failure characteristics are not identified by the data base. Uncertainties in data base results can be addressed by formal uncertainty analyses or by sensitivity studies.

Once it is established that aging has been affecting component reliability, system availability or plant safety, corrective measures at the design, operating, and maintenance level are considered for mitigating aging effects. Since altering design or operational procedures are difficult to implement unless it is absolutely necessary, in most cases, the maintenance, surveillance or other catalytic factors shown in Figure 1 are enhanced. Thus, the data bases not only help understand the aging effects, but also are valuable for identifying the type and level of effort necessary for improving plant performance.

In conclusion, the data bases serve a very important role in understanding aging in nuclear power plants. Standardization of the data entry process would eliminate some of the cumbersome review process currently required. In addition, making the data more accessible would enhance its usefulness as an analytical tool.

References

1. MIL-HDBK-217B, Reliability Prediction of Electronic Equipment, Washington, DC, Department of Defense, September 1974.

2. NUREG-1144, Rev. 1, Nuclear Plant Aging Research (NPAR), Program Plan, USNRC, September 1987.

3. B. M. Meale and D.G. Satterwhite, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," INEL NUREG/CR-4747, December 1986.

4. W.L. Greenstreet, G.A. Murphy, and D.M. Eissenberg, "Aging and Service Wear of Electric Motor-Operated Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants," ORNL, NUREG/CR-4234, July 1985.

5. K.R. Hoopingarner, et al., "Aging of Nuclear Station Diesel Generators: Evaluation of Operating and Expert Experience," PNL, NUREG/CR-4590, Vols. 1 and 2, August 1987.

6. W. Shier and M. Subudhi, "Operating Experience and Aging Assessment of Motor Control Centers," BNL, NUREG/CR-5053, July 1988.

7. W.E. Gunther, R. Lewis, and M. Subudhi, "Detecting and Mitigating Battery Charger and Inverter Aging," BNL, NUREG/CR-5051 (In print).

8. J. Higgins, et al., "Operating Experience and Aging Assessment of Component Cooling Water Systems in Pressurized Water Reactors," BNL, NUREG/CR-5052, July 1988.

DEGRADATION OF SA533B Cl.1 STEEL DUE TO ACCELERATED THERMAL AGING

K. Iida, K. Miya, Y. Asada, A. Okamoto
Y. Kitsunai, N. Ohtsuka and N. Urabe

Summary

In 1987, the Life Extension (LE) Research Committee was organized in the Japan Welding Engineering Society to execute experimental research related to the life extension for the nuclear power plants. As the first stage work of the four-year research program, the Committee has carried out a literature survey of domestic and foreign papers concerned in order to address specific research objectives. The reactor pressure vessel for light water reactors was, consequently, focused for the research object and the SA533B Cl.1 steel has been manufactured and rolled to a 150 mm thick test plate. The chemical composition of the test plate was intentionally adjusted referring to the reactor pressure vessel steels for the Japanese older commercial nuclear power plants.

The test plate is now subjected to accelerated aging treatments, those are iso-thermal aging, step-cool aging and strain-thermal aging. Succeeding to the aging treatments, mechanical tests to evaluate the degree of degradation, physical and metallographic examinations to identify the deterioration mechanisms and trials of application of non-destructive tests to detect the degradation are performed.

Introduction

Thirty-five commercial nuclear power plants are under services in Japan as of July 1988. The light water reactor (LWR) systems are expected to occupy a key position in the electricity supplier plants for coming several decades. In 1986, Advisory Committee for Energy - a consultive body of the Ministry of forecast regarding and Industry (MITI) - issued a vision regarding the nuclear power generation in the 21st century. Taking a good economy balance into consideration, the life extension of the existing nuclear power plants is concluded as one of the strategies, and thus the tactics are proposed to develop the technical schemes for the life extension.

According to this proposal, several research programs have been planned and now carrying out in Japan. Those are described briefly as follows.

The Japan Power Engineering and Inspection Corporation (JAPEIC) started in 1985, under the sponsorship of the MITI, an eight-year experimental work of Technical Development Program for Nuclear Power Plant Life Extension[1] in co-operation with three nuclear plant manufacturing companies. The main programs are low cycle fatigue tests of low-alloy steels and stainless steels under light water reactor environment, fracture toughness tests and stress corrosion cracking tests of thermal-aged and neutron irradiated stainless steels. A development of monitoring system and inspection, repairing and replacement techniques is also included.

The Joint Research Project by the electric power supplier companies has finished two-year feasibility study regarding the plant life extension and started in 1987 the specific tasks to develop the monitoring system for fatigue phenomena during the course of transient operation of the nuclear power plant reactors.

The Japan Atomic Energy Research Institute[2,3] and the Central Research Institute of Electric Power Industry[4] are also conducting individually the studies related to the plant life extension.

Under these circumstances, the Life Extension (LE) Research Committee was organized at the beginning of fiscal year 1987 in the Japan Welding Engineering Society under a contract of the Japan Atomic Energy Research Institute, in order to carry out a four-year research program related to the life extension of light water nuclear power plants. More than fifty researcher members, selected from universities, research institutes and industry companies are participating in the committee and progressing the activities. For the first year task, the literature survey has been carried out to address specific research subjects for the life evaluation of the nuclear power plants. The reactor pressure vessel for the light water reactor system was, consequently, selected for the research object and a SA533 Gr.B Cl.1 steel was manufactured and rolled to a 150 mm thick test plate to investigate factors and mechanisms of potential degradation. As the chemical compositions of the reactor pressure vessel steels of the Japanese older commercial LWR system were referred, the chemical composition of the test plate was intentionally controlled to enhance the deterioration effects of impurity atoms. The aging conditions were also decided to accelerate the degradation of the test plate.

In this paper, a summary of the literature survey, the results of mechanical properties of the test plate in as-received state and the scope of the research program of the LE Research Committee are presented.

Literature Survey on Material Degradation Studies

The material degradation phenomena postulated to occur in the nuclear power plant reactor pressure vessels and structural components were surveyed in the first year task of the LE Research Committee. The materials examined in those studies were SA533B Cl.1 steels,[5,6,7,11,12,13,14] SA508 Cl.2 and Cl.3 steels,[6,7,12,13] low carbon steels,[8,9] Ni based alloys,[8] CrMoV steels,[9] stainless steels,[8,9] 3.5 NiCrMoV steels,[10] 2¼Cr-1Mo steels[12] and weldments of these steels. Those materials were given accelerated degradations either by thermal aging, strain-thermal aging or neutron irradiation.

The degrees of material degradation were evaluated by the strength or the ductility of the tensile properties, the ductile-brittle transition temperature or the upper shelf energy of Charpy impact properties, or the fatigue crack propagation rate or the fracture toughness and so on. The causes of the degradation were investigated by the electron micrography or Auger electron spectroscopy. The mechanisms of the thermal aging and the strain-thermal aging were explained mainly by the grain boundary segregation of impurity atoms (P, Sn, Sb, As.....) or migration of carbon or nitrogen atoms at around discontinuities. The precipitation of carbides and intermetallic compounds also contribute the material degradation, thus, the transition temperature shifts to higher temperature and the upper shelf energy decreases in value.

The studies, which have been reported in the literature, for the thermal aged or the strain-thermal aged MnMoNi steels are summarized in Table 1. Material 1[5] and 3[7,12,13] are SA533B steels and material 6[7,12,13] is SA508 steel, material 4, 5 and 7[7,12,13] are simulated

Table 1 Thermal and Strain-thermal Aged MnMoNi Steels in Literature

Material #	Material	Chemical Composition (wt%) C	Si	Mn	Ni	Cu	P	S	Sn	Sb	As	Accelerated Aging	Ref.
1	SA533B - 165mm thick plate, PWHT for RPV	0.18	0.22	1.42	0.63	0.04	0.005	0.006	0.006	0.003	0.005	Iso-thermal Aging (350-450C)x (8500-17500hr)	(5)
2	SA533B - Weld Metal, 620C x 30hr PWHT	0.07	0.16	1.52	1.56	0.07	0.008	0.011	0.005	-	0.010		
3	SA533B - 150mmt Plate, 605/622C x 24hr PWHT	0.21	0.22	1.43	0.59	0.11	0.005	0.005	0.030	<0.010	0.020	Iso-thermal Aging (300-500C)x (100-20000hr)	(7)(12)(13)
4	SA533B - Sim. C. G. HAZ, 615C x 25hr PWHT	-	-	-	-	-	-	-	-	-	-		
5	SA533B - Sim. F. G. HAZ, 615C x 25hr PWHT	-	-	-	-	-	-	-	-	-	-		
6	SA508-Cl3-545mmt Flange, 600/625C PWHT	0.23	0.29	1.49	0.79	0.05	0.008	0.004	<0.010	<0.010	<0.010		
7	SA508 - Sim. C. G. HAZ, 600/625C PWHT	-	-	-	-	-	-	-	-	-	-		
8	SA533B - 30x75mm Forged Bar, "Pure"	0.24	0.26	1.37	0.54	0.04	0.0027	0.0047	<0.010	<0.010	<0.020	Iso-thermal Aging (500C)x (1000hr)	(14)
9	SA533B - 30x75mm Forged Bar, "S-doped"	0.25	0.27	1.47	0.60	0.09	0.0025	0.0610	<0.010	<0.010	<0.020		
10	SA533B - 30x75mm Forged Bar, "Cu-doped"	0.27	0.28	1.39	0.55	0.40	0.0086	0.0053	<0.010	<0.010	<0.020		
11	SA533B - 30x75mm Forged Bar, "P-doped"	0.26	0.29	1.44	0.54	<0.01	0.0450	0.0045	<0.010	<0.010	<0.020		
12	SA533B, Low C-high N content	0.16	0.24	1.48	0.58	0.12	0.009	0.12	-	-	-	Strain-thermal Aging (2.6%)x(300C)x (100hr)	(6)
13	SA533B, Ordinary C, N content	0.21	0.25	1.38	0.66	0.12	0.011	0.010	-	-	-		
14	SA508-Cl2, High N content	0.24	0.21	0.78	0.87	0.09	0.004	0.011	-	-	-		
15	SA533B	0.19	0.25	1.49	0.56	0.13	0.014	0.013	-	-	-	Strain-thermal Aging (180/390MPa)x (100/288C)x (48 35000hr)	(11)

heat affected zones for each steel, and material 2[5] is a weld metal suited for SA533B steel. For these materials, the changes in ductile-brittle transition temperature caused by the thermal aging treatments have been investigated as a function of the aging temperature and the aging time. Figure 1 shows the ranges of aging conditions in Table 1, namely the aging temperature vs. the aging duration. The parameter TP in the figure is the temper parameter given in eq.(1)

$$TP = (C + \log t) \times 10^{-3} \qquad (1)$$

where T is the absolute temperature at the aging and t is the time in hours for the aging duration and the constant C is evaluated as 10 using modified Hollomon-Jaffe equation. The relationship between the change in the transition temperature and the parameter TP to evaluate the degree of material degradation due to thermal aging is plotted in Fig. 2. The change in the transition temperature increases with increase in the TP, though the trend differs by each material. The largest change in transition temperature is recognized in the material 6, but no change has occurred at all in the material 1. This difference might be due to the fact that the contents of phosphorus and silicon are higher than those in the material 1. If the material 2 and 6 are compared, the silicon content is higher in the material 6, but nickel content is higher in the materials 2, while the phosphorus content is the same in both materials. Thus the effect of silicon seems larger than the nickel on the material degradation. On the other hand, the materials 4, 5 and 7 are the simulated heat affected zones with different grain sizes. The larger grain sized HAZ shows the larger change in the transition temperature. Thus the grain size may be another important factor of the material degradation. In the materials from 8 to 11[14], the effect of impurity atom contents has been studied and it was reported that the most deteriorating impurity atom is phosphorus for thermal aging rather than sulfur or copper atoms. The straining prior to the thermal aging seems to cause additional degradation into the material.[6,11]

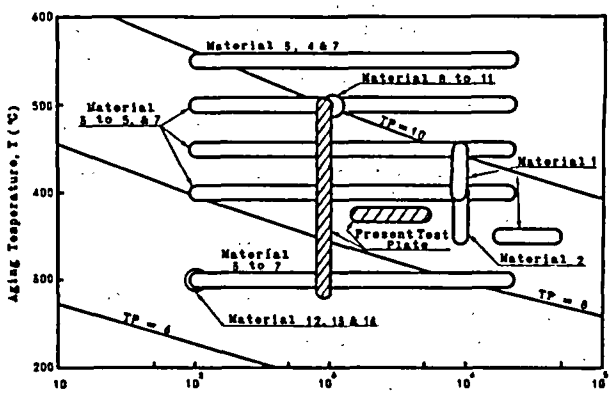


Fig. 1 Comparison of Degradation Conditions in the Relationship between Aging Temperature and Duration

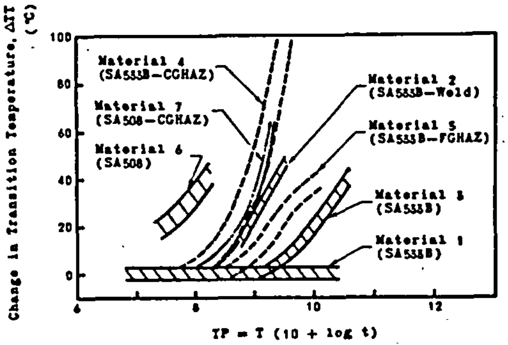


Fig. 2 Relationship between Change in Transition Temperature (ΔTT) and Temper Parameter (TP) for MnMoNi Steels in Literature

Using literature data in Table 1, the effect of impurity atoms was investigated on the degradation phenomena due to the thermal aging. As the results, a considerable role of the phosphorus on the material degradation was recognized, while a relatively small effect of the silicon and little effect of the sulfur were realized. For an example, the relationship between the change in the transition temperature and the content of phosphorus is shown in Fig. 3. Although, there is a wide scatter in it, a linear relationship

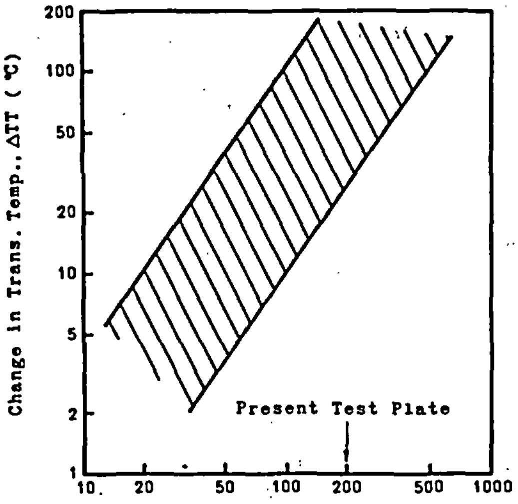


Fig. 3 Relationship between ΔTT and Phosphorus Content

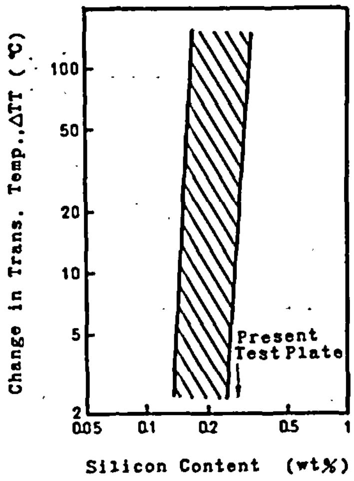


Fig. 4 Relationship between ΔTT and Silicon Content

exists between them. Figure 4 indicates the similar example on the relationship of the change in transition temperature with the content of silicon. Comparison of Fig. 4 with Fig. 3 supports the above mentioned results that the contribution of the silicon to the aging degradation is smaller than the phosphorus. Thus, it is concluded that the impurity atom plays different roles in the thermal aging for each other and its effect closely related to the temper parameter in eq. (1), the grain size of material and the amount of strain. Much more data based on systematic experiments are required to understand the material degradation due to the aging.

Program of the LE Research Committee

Material

The material selected is a SA533B Cl.1 steel according to the discussion in the previous section. A 150 mm thick plate has been rolled and heat-treated as follows,

Normalizing at 850 - 925°C for 7 hours
Tempering at 640 - 665°C for 6.3 hours
Quenching at 860 - 895°C for 5.5 hours
Tempering at 650 - 675°C for 4.8 hours
then
Post Weld Heat Treatment at 605 - 625°C for 46 hours.

In Table 2, the results of chemical analysis and the standard values for the SA533B steel are shown. The contents of impurity atoms such as phosphorus, sulfur and copper are higher than those of modern commercial SA533B steels. For the comparison's sake, the contents of phosphorus and silicon of the test plate are shown by arrows in Fig. 3 and 4. The mechanical test results of the test plate in as-received state are given in Table 3 and the transition behaviours for the absorbed energy, the lateral expansion and percent of shear fracture by the Charpy impact test are also shown in Fig. 5.

Degradation Treatment

The test plate is now subjected to various aging treatments. Those are,

1) Iso-thermal aging at 288, 350, 375, 400, 450 and 500°C for 35, 66 or 200 days. The calculated temper parameter TP is ranged between 7.26 and 10.00.
2) Step-cool aging as
593°C x 1hr 538°C x 15hr 524°C x 24hr 496°C x 48hr finally 468°C x 72hr.
3) Strain-thermal aging both for straining prior to the thermal aging and the thermal aging under the strain.

The iso-thermal aging conditions are shown in Fig. 1 in order to compare with those given in Table 1.

Table 2 Chemical Composition of the Present Test Plate (wt%)

Element	C	Si	Mn	P	S	Ni	Cr	Mo	Cu	V	Al	Sn	As	Sb
Standard	<0.25	0.13/0.32	1.10/1.55	<0.035	<0.040	0.37/0.73	-	0.41/0.60	-	-				
Top	0.21	0.28	1.47	0.020	0.014	0.61	0.14	0.50	0.16	<0.010	0.023	0.010	0.014	0.0096
Bottom	0.19	0.27	1.45	0.020	0.012	0.60	0.14	0.51	0.16	<0.010	0.013	0.011	0.014	0.0085

Table 3 Mechanical Properties of the Present Test Plate (after PWHT)

Tests / Location	Tensile Test				Charpy Impact Test				Drop Weight Test	
	Proof Stress (MPa)	Tensile Strength (MPa)	Elongation (%)	Reduction in Area (%)	DBTT (°C)	TT_{41J} (°C)	TT_{68J} (°C)	USE (J)	T_{NDT} (°C)	RT_{NDT} (°C)
Top	488	642	27.8	65.0	-6	-20	-8	167	-25	-25
Bottom	472	620	27.6	65.4						

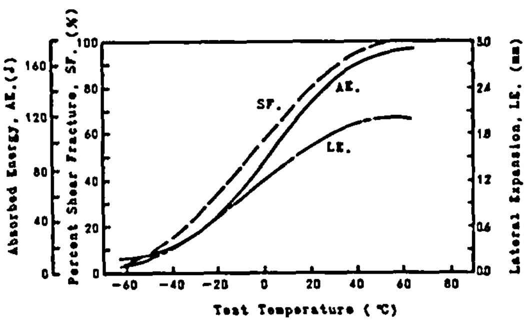


Fig. 5 Charpy Impact Transition Curve of the Present Test Material

Program of Tests

Succeeding to the aging treatments, tensile tests at RT, 100, 200, 300, 350, 375, and 400°C varying the strain rate, Charpy impact tests, drop weight tests, fracture toughness tests, fatigue tests both in air and corrosive environment are planned. Physical and metallographical tests and non-destructive tests will be performed.

Similar kind of tests are also carried out on the virgin state of the test plate. The matrix of the experiments is summarized in Table 4.

In the fracture toughness tests on the strain-thermal aged material, the effect of warm pre-stressing on the fracture toughness is planned to be performed. Non-destructive tests such as eddy current technique for the aging degradation will be conducted.

Concluding Remarks

For the first year task, the LE Research Committee summarized the aging phenomena on the MnMoNi steel using literature data. The degradation due to the impurity atom segregation may be explained by the contents of impurity atoms and by the temper parameter TP. The SA533B Cl.1 plate of 150 mm thick has been manufactured and is subjected to the iso-thermal aging, the step-cool aging and the strain-thermal aging. The phosphorus content is 200 ppm in the test plate and more than 30°C shift in the transition temperature can be expected (cf. Fig. 3).

Acknowledgements

The authors wish to thank the members of the LE Research Committee for their elaborated works to finish the first year task, and the wishes are also extended to the financial support by the Japan Atomic Energy Research Institute.

References

1. Y. Mishima, Light Water Reactor Power Plant Life Extension Program in Japan, IAEA-SM-295/45, Int. Sym. on Safety Aspects of the Ageing and Maintenance on Nuclear Power Plants, Vienna, 1987.
2. H. Nakajima, T. Shoji, N. Nakajima, H. Takahashi and T. Kondo, A Material Engineering Aspect of Aging Phenomena in Structural Materials of LWR and their Relevance to R and D in Plant Life Extension programme, IAEA-295/44, ibid, 1987.
3. S. Miyazono, Life Assessment Study of Light Water Reactor Structural Components in Japan, OECD/NEA/CSNI-UNIPEDE/NUCLE Specialist Meeting on Life Limiting and Regulatory Aspects of Reactor Core Internals and Pressure Vessels, Stockholm, 1987.
4. Nuclear Industry News (Genshiryoku-sangyo-sinbun), August 6, 1987 (in Japanese).
5. R. Pelli and J. Forsten, Effect of Thermal Ageing on Impact Ductility of The Nuclear Pressure Vessel Steel SA 533B and Its Weld Metal, IAEA Specialists' Meeting on "Load and Time Dependent Material Performance other than Irradiation" Budapest, 1986.
6. P. Pelli, K. Torronen, S. Salonen and K. Rahka, Strain Ageing of Nuclear Pressure Vessel Steels A533B and A508 Cl.2, IAEA Technical Meeting on "Time and Load Dependent, Deg. Pressure Boundary Materials," Innsbruck, 1978.

Table 4 Matrix of the Present Test Program

Tests	Aging Condition	Iso-thermal Aging								Step Cool Aging	Strain Thermal Aging	Remarks
	T(°C)	-	288	350	375	400	375	450	500			
	Time(hr)	-	864	864	1584	864	4800	864	864			
	TP*	-	7.26	8.06	8.55	8.71	8.87	9.35	10.00			
Tensile Test	RT	o	o	o	o	o	o	o	o	o		Strain Rate at 10^{-5}, 10^{-4}, 10^{-3}/sec
	100°C	o			o		o					
	200	o			o		o					
	300	o	o	o	o	o	o	o	o	o		
	350	o			o		o					
	375	o			o		o					
	400	o			o		o					
Charpy Impact Test		o	o	o	o	o	o	o	o	o		TT, USE, LE
Drop Weight Test		o			o		o					Single Bead
Fract. Tough. Test		o	o	o	o	o	o	o	o	o	o	Static and Dynamic
Fatigue Test		o			o		o					in Air and Corr. Envir.
Phys. Metall. Test		o	o	o	o	o	o	o	o	o	o	AES, EMF
Nondestructive Test		o			o		o					Eddy Current, UT, AE

* Temper parameter given in eq.(1)

7. S.G. Druce, G. Gage, G.R. Jordan and J.A Hudson, Effect of Thermal Ageing on Mechanical Properties of PWR PV Steels and Weldments, 8th SMiRT Conf. on Structural Mechanics in React. Tech., Brussels, 1985, p401.
8. L.A. James, Effect of Irradiation and Thermal Ageing upon Fatigue-Crack Growth Behavior of Reactor Pressure Boundary Materials, IAEA Technical Meeting on "Time and Load Dependent Deg. Pressure Boundary Materials," Innsbruck, 1978.
9. J. Gisztler, Thermal Strain Ageing and Crack Initiation During Low Cycle Thermal Shock Fatigue, IAEA Specialists' Meeting On "Load and Time Dependent Material Performance Other Than Irradiation," Budapest, 1986.
10. J.R. Gorden and W.J. Skelton, The Effect of Temper Embrittlement on the Fracture Properties of A Pressure Vessel Steel, 8th SMiRT, F2 1/4, August 19-23, 1985 Brussels, Belgium.
11. O. Sandberg, Influence of Strain Ageing on Fracture Toughness of The Pressure Vessel Steel A533B at Elevated Temperatures, 8th SMiRT, F2 1/2, August 19-23, 1985 Brussels, Belgium.
12. J.A. Hudson, S.G. Druce, G. Gage and M. Wall, Thermal Ageing Effects in Structural Steels, IAEA Specialists' Meeting on "Load and Time Dependent Material Performance other than Irradiation," Budapest, 1986.
13. S.G. Druce, G. Gage and G. Jordan, Effect of Ageing on Properties of Pressure Vessel Steels, Acta Metall. Vol. 34 (1986), pp 641-653.
14. S.G. Druce, Effects of Austenitisation Heat Treatment on the Fracture Resistance and Temper Embrittlement of MnMoNi Steels, Acta Metall. Vol. 34 (1986), pp 219-232.

AGING ASSESSMENT OF LWR COOLANT PUMPS[a]

Vikram N. Shah
William L. Server
Philip E. MacDonald

Abstract

This paper summarizes some of the results of the Aging Assessment and Mitigation Project sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The objective of the project is to understand and manage aging degradation of the major light water reactor (LWR) structures and components, so that the impact of aging on the safe operation of the nuclear power plants can be evaluated and addressed. This objective is being accomplished by integrating, evaluating, and updating the available aging-related information. This paper discusses current accomplishments and summarizes the significant degradation processes active in the pressurized water reactor primary coolant pumps. It evaluates the effectiveness of the current in-service inspection programs to detect degradation damage and, finally, presents conclusions and recommendations related to aging of the LWR coolant pumps.

Introduction

One hundred nine commercial nuclear power plants are currently licensed to operate in the United States. The oldest PWR is the Yankee Rowe plant, which has been in operation for 28 years. Five other plants have been in operation for more than 20 years. Twenty-two plants have been in operation between 15 and 20 years. Several time-dependent degradation mechanisms not accounted for in the original design have caused failures in the field. Some of the recent aging-related problems include a steam generator tube rupture caused by high-cycle fatigue, significant wall thinning of a metal containment caused by corrosion, catastrophic failure of a "nonnuclear" portion of a pressurized water reactor (PWR) feedwater line caused by erosion-corrosion, and a through-wall crack in a PWR safety injection pipe between the nozzle and first check valve caused by thermal fatigue.

Therefore, the potential problems of understanding and managing aging have become a major focus for the research sponsored by the U.S. Nuclear Regulatory Commission (USNRC). An important part of the USNRC research effort is the Nuclear Plant Aging Research (NPAR) Program that is being conducted at several national laboratories, including the Idaho National Engineering Laboratory (INEL).[1]

The NPAR Program is sponsoring the Aging Assessment and Mitigation Project at the INEL to understand aging of the major light water reactor (LWR) structures and components and to identify the technical issues associated with aging. The major objective of this project is to integrate, evaluate, and update the current information related to time-dependent degradation or aging of the major LWR components so that the technical issues that may impact safety can be identified and addressed in an effective and timely manner. This information will provide the basis for the development of life assessment procedures, which will assist the USNRC in the formulation of license renewal policy, as well as other regulatory applications.

A five-step approach is being followed to accomplish the project objectives.[2] This approach includes the following:

1. Identify and prioritize the major reactor components according to their relevance to plant safety.
2. Identify the degradation sites, mechanisms, stressors, and potential failure modes.
3. Assess current and emerging inspection, surveillance, and monitoring methods to detect, quantify, and trend aging degradation. Evaluate current and emerging maintenance methods to mitigate aging damage.
4. Develop life assessment procedures.
5. Develop criteria for license renewal.

The first two steps of the five-step approach have been completed, and some progress has been made on the remaining three steps. This paper summarizes the qualitative results of aging assessment for the PWR primary coolant pump and the BWR recirculation pump.[2,3] It includes description of pump design and fabrication, analysis of active degradation processes, and assessment of current in-service inspection programs. Finally, it presents the conclusions and recommendations related to aging degradation of LWR coolant pumps.

Selection of Major LWR Components

The criteria employed to select and prioritize the major components are based on safety; the components are those that will help contain or mitigate any release of fission products that may take place

---

[a] Work sponsored by the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570.

during normal operation, off-normal conditions, design-basis accidents, or a severe accident. The main concern is that the aging may reduce the safety margins of the major components. The major components selected for study are listed in Table 1. The safety-related criteria and engineering judgment are used to rank these components.[2,4] These components include primary pressure boundary components, primary containments, reactor pressure vessel supports, reactor internals, emergency diesel generators, cables and connectors, and feedwater and steam lines.

TABLE 1. SELECTED MAJOR PWR AND BWR COMPONENTS

Major PWR Components

1. Reactor pressure vessel (RPV)
2. Containment and basemat
3. Reactor coolant piping, safe ends, and nozzles
4. Steam generators
5. Reactor coolant pump body
6. Pressurizer and surge and spray lines
7. Control rod drive mechanisms
8. Cables and connectors
9. Emergency diesel generators
10. RPV internals
11. RPV supports
12. Feedwater lines and nozzles

Major BWR Components

1. Containment and basemat
2. Reactor pressure vessel
3. Recirculation piping and safe ends
4. Main steam and feedwater lines
5. Recirculation pump body
6. Control rod drive mechanisms
7. Cables and connectors
8. Emergency diesel generators
9. RPV internals
10. RPV supports

The reactor pressure vessel has been identified as the most critical major component in PWR plants. The catastrophic failure of a reactor pressure vessel could lead to rapid core meltdown, generating high-pressure and temperature loadings for which PWR containments are not designed. Therefore, it is ranked higher than the other PWR components. The containment and basemat have been identified as the most critical major components in BWR plants, because they act as a barrier, protecting the public from released fission products during a severe accident.

The Aging Mitigation Project has identified the degradation sites, stressors, and degradation mechanisms for each of the major components. In addition, the project has discussed the potential failure modes for the aged components and the current inspection methods used to detect aging damage. Irradiation embrittlement, mechanical and thermal fatigue, intergranular and transgranular stress corrosion cracking, thermal embrittlement, erosion, corrosion, and fretting have been identified as some of the important degradation mechanisms active in the major components. The results for the aging of both PWR and BWR reactor coolant pumps are summarized here.[5]

Aging Assessment of LWR Coolant Pumps

The primary function of the LWR coolant pumps is to circulate coolant through the reactor such that core-heated fluid can be passed to a turbine (in BWR plants) or to a steam generator (in PWR plants). The reactor coolant pumps are the only rotating heavy machines on the nuclear side of a plant and are critical pressure boundary elements. This section discusses the aging degradation of three pump components: pump casing, closure studs, and pump shaft. The pump shaft seals are not addressed because they are replaced or refurbished relatively frequently. First, the design and fabrication of the LWR coolant pumps are discussed. Then, the degradation mechanisms for the pump components are assessed. Finally, the inspection methods for the pump components are evaluated.

Reactor Coolant Pump Design and Fabrication

The design of the reactor coolant pumps fall into one of three categories or types, as shown in Table 2: PWR primary coolant pumps, i.e., Types F and E; and BWR recirculation pumps, i.e., Type C. The major difference in the reactor coolant pump designs is in the pump body geometry. The PWR Type F pump casings require thicker pressure retaining walls (4 to 8 in.) because the casing walls are not structurally supported, as shown in Figure 1. The PWR Type E and BWR Type C pump bodies are shown in Figures 2 and 3, respectively; their casing walls are thinner (2.5 to 3 in.) because of the structural reinforcements in their designs. The massive size of the pump casing is required to maintain the close tolerances required by the pump internals during operation. Recirculation pumps in BWR plants are smaller than the primary coolant pumps in PWR plants owing to lower capacity, i.e., a smaller amount of liquid being moved. In addition, less space is available in a BWR containment.

TABLE 2. MANUFACTURERS AND NSSS VENDORS FOR REACTOR COOLANT PUMPS

Pump Type	Manufacturer	NSSS Vendor
Type E	Byron-Jackson	Babcock & Wilcox (3 plants) Combustion Engineering
Type F	Westinghouse	Westinghouse (all plants) Babcock & Wilcox (2 plants)
	Bingham-Willamette	Babcock & Wilcox (2 plants)
	Combustion Engineering/ Klein, Schanzlin, and Becker	Combustion Engineering (Palo Verde Plants)
Type C	Byron-Jackson Bingham-Willamette	General Electric

All reactor coolant pumps are fabricated from statically cast austenitic stainless steel, either Grades CF8, CF8M, or CF8A, except the ones at the three units of the Palo Verde plant. The pump casings at Palo Verde are fabricated from forged carbon steel and clad with stainless steel. The sections of Type F pump casing made of cast stainless steel are welded by an electroslag method, which introduces high residual stresses that may exceed the yield strength level of the material. However, no post-weld heat treatment is performed to reduce or eliminate the residual stresses at the fabrication welds in Type F pumps; a post-weld

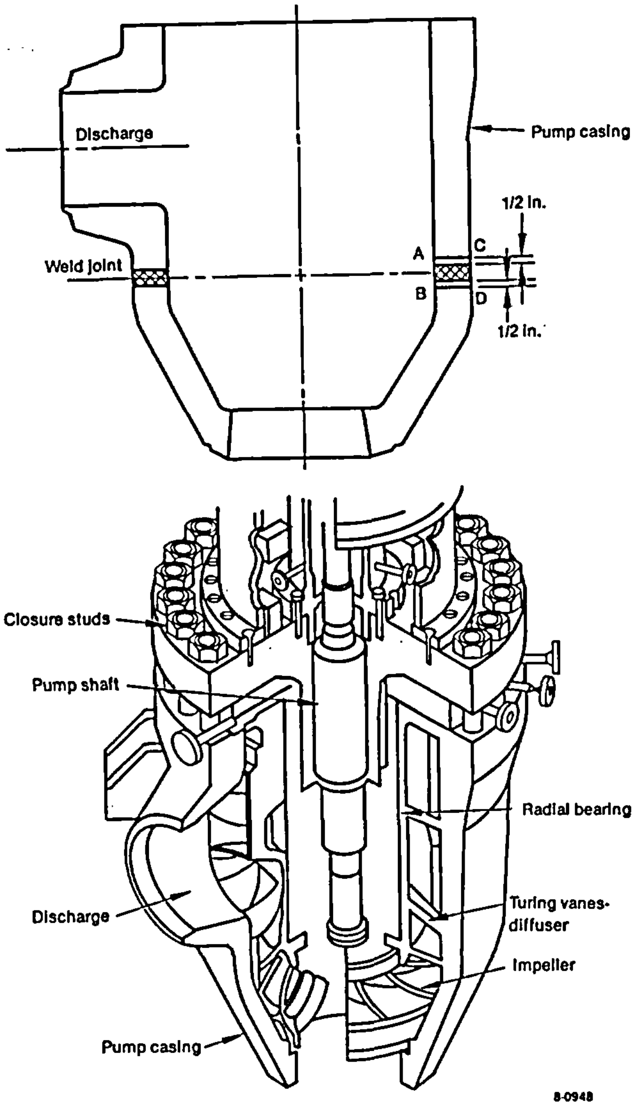


Figure 1. PWR Type F coolant pump.

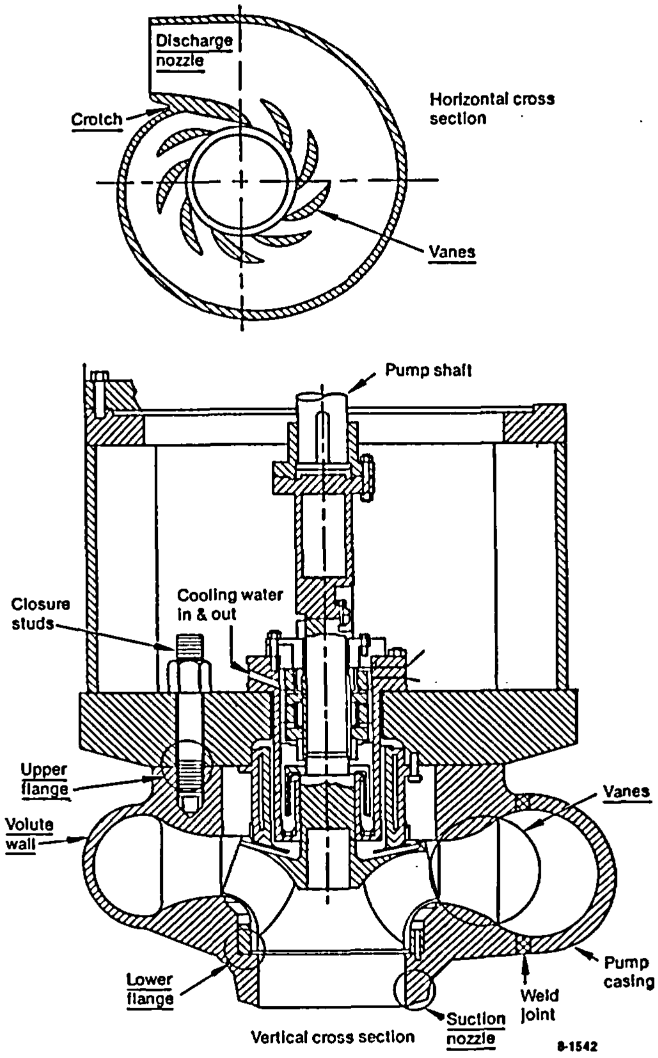


Figure 2. PWR Type E coolant pump (locations of maximum stress intensity are underlined).

heat treatment is not an ASME requirement for a stainless steel weld. Later designs of Type F pump casings were modified to decrease or eliminate the number of weldments, as it became practical to manufacture larger casting sections.[6] The carbon steel pump casings are subjected to post-weld heat treatment, and all Type E and Type C pump casings are subjected to full-solution heat treatment, which will eliminate almost all residual stresses.

A thermal barrier or heat exchanger, or both, are used to limit the reactor coolant heat reaching the mechanical seal cavity. In the earlier Types E and C pumps, the hot reactor coolant was mixed with the cold cooling water at the top of the thermal barrier. The resulting turbulent mixing introduced high-cycle (1 to 25 Hz) thermal fatigue loads on the pump shaft surface.

Two concentric 304 SS flexitallic (stainless steel-graphite asbestos material) gaskets are used for sealing between the PWR coolant pump cover and casing. A leak-off line is installed between the gaskets to detect any leakage of reactor coolant. Only one gasket is used in a BWR coolant pump. The leaking reactor coolant, if not checked, may cause corrosion of the closure studs, which are made of low alloy steel, i.e., SA193 Grade B7 or SA540 Grade B23. Reactor coolant pump main closure studs are long, e.g., 29-in.-long studs for Type E pumps,[7] and their nominal diameter may vary from 3.5 to 5.5 in.

Aging of Reactor Coolant Pumps

Thermal Embrittlement. Thermal embrittlement is a major degradation mechanism for the cast stainless steel reactor coolant pump casings. Cast stainless steels, i.e., Grades CF8, CF8M, and CF8A, have austenitic-ferritic microstructures and are subject to thermal embrittlement with prolonged exposures at the operating temperature of 288°C (550°F). Thermal embrittlement of the base metal results in a slow loss of material toughness over extended periods of time and is influenced by coolant temperature, time of exposure at temperature, chemical composition, volume fraction of ferrite content, and ferrite distribution (spacing) in the microstructure. Larger ferrite content and spacing will cause increased thermal embrittlement. A high percentage of ferrite (18 to 22% is not uncommon) can be present in the cast stainless steel components. In thick-section stainless steel casting, such as pump casing, the ferrite spacing through the wall is not uniform.[8] The loss of material toughness is caused by the formation of alpha-prime and G phases in the ferrite. As only the ferrite phase is embrittled by long-term exposure at LWR operating temperatures, the overall

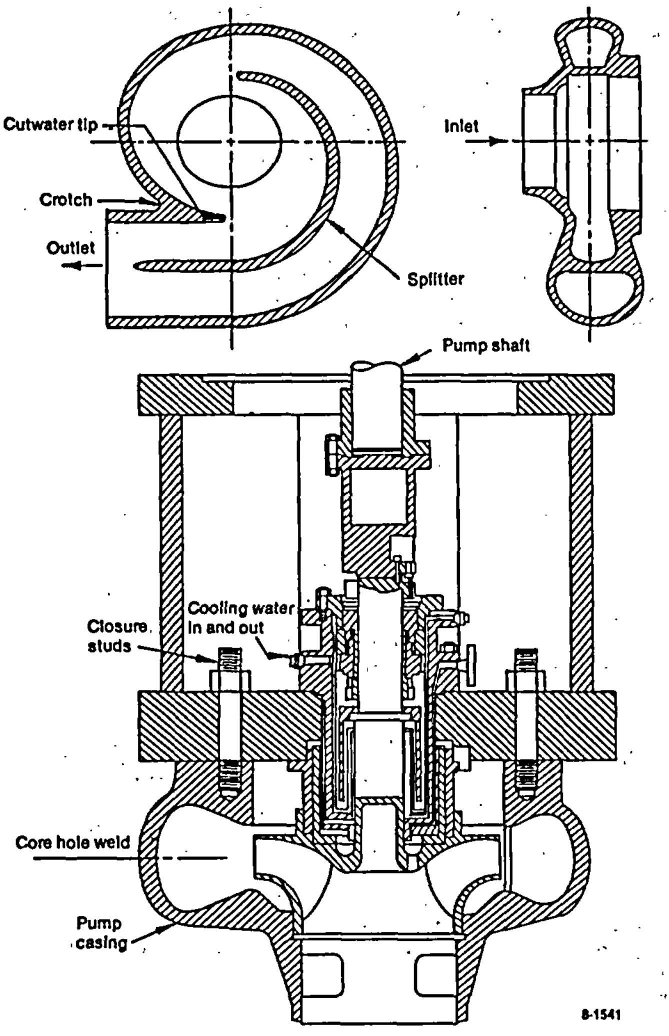


Figure 3. BWR Type C coolant pump.

embrittlement of pump casing depends on the amount and spacing of ferrite present in the base metal of the pump casing.

The weldments in the pump casing are not very sensitive to thermal embrittlement. The initial toughness of the weldments in the pump casing is significantly lower than that of the unaged base metal, and the ferrite content in the weldments is generally lower than that in the base metal. In several older plants, the weldments in the pump casing have less than 3% of ferrite,[9,10] whereas in newer plants a minimum of 5% of ferrite is required.[5,9] These requirements for minimum ferrite avoid microfissuring in weldments.

Some laboratory tests show that the loss in toughness of thermally aged cast stainless steel components can be recovered by annealing at 550°C for one hour, followed by rapid cooling, i.e., quenching, to lower temperatures.[11,12] The short-term annealing process dissolves the alpha-prime phase. However, rapid heating and cooling of the pump casing is not feasible, and slow heating and cooling to and from the annealing temperature will cause formation of several other phases in the ferrite, resulting in additional loss of toughness. Therefore, annealing of the aged cast stainless steel components is not an acceptable solution.

The actual degree of thermal embrittlement that has occurred in cast stainless steel components exposed to the LWR environment needs to be estimated to assess the structural integrity of these components. Thermal embrittlement may be estimated by analytical modeling of in-service degradation, nondestructive examination of components, and testing of small samples removed from the components for mechanical properties, i.e., tensile strength, hardness, and fracture toughness.[13] The analytical models for in-service degradation are based on the room temperature Charpy-impact energy data for the specimens aged at 300 to 400°C. Development of a model to estimate degradation in fracture toughness rather than impact energy is recommended. Nondestructive examination, i.e., ultrasonic testing, of components can be used to estimate the degree of thermal embrittlement, if a correlation between ultrasonic attenuation and the fracture toughness of the aged cast stainless steel component can be developed.

Fatigue. Pump casing is subjected to thermal and mechanical fatigue damage caused by the system operating transients and pump vibrations. In Type F pumps, the weldments are susceptible to fatigue damage because of the high residual stresses, whereas in Types C and E pumps the susceptible sites are likely to include some portion of both the base metal and weld region, because of the different geometric configuration and complex welds with no residual stresses. Figure 2 shows the locations of maximum stress intensities in a Type E pump.[6] In addition, in the older plants the presence of microfissures in the weldments having a lower ferrite content (<3%) may adversely affect the fatigue strength of the pump casing.[10] However, the fatigue damage is expected to be quite small in the absence of microfissures, because the pump casings have a thickness greater than that required for structural integrity to ensure dimensional stability.

The potential flaws in the cast stainless steel pump casing are introduced during fabrication, and result from shrinkage during solidification. If these flaws are located on the interior surface of the pump casing, they will be exposed to the reactor coolant and may be subject to crack growth rates considerably greater than those for the subsurface flaws.[6]

The pump shafts are also susceptible to fatigue damage caused by the alternating bending stresses from asymmetric distribution of static pressure and by the rapidly varying thermal stresses from the turbulent mixing of hot reactor coolant (∿550°F) with cooling water (∿125°F) in the cover thermal barrier. Thermal stresses from turbulent mixing, alternating bending stresses, and high residual stresses at the local welds are responsible for the initiation and propagation of fatigue crack. Axial and circumferential cracks have been found on some PWR and BWR pump shafts.[14,15] Axial cracks are caused by the turbulent mixing of the coolants, whereas the circumferential cracks are caused by the alternating bending stresses and located in the grooves on the shaft surface. Hairline cracks have been found in the Type F pump shafts at the Palo Verde-1 plant. Heat-induced stress and the shaft's chrome plating have been mentioned as factors causing the cracks.[16]

Boric Acid Corrosion. PWR pump body closure studs are susceptible to corrosion by borated primary coolant leakage across the pump casing-to-cover gasket. Such leakage has caused significant corrosion damage to the studs, and in carbon steel pump bodies it may also cause corrosion of the base metal if exposed to leaking coolant.[17] Boric acid corrosion of the studs will increase the rate of leakage which, in turn, will lead to excessive corrosion of studs. In one PWR plant, boric acid corrosion reduced seven reactor coolant pump studs from a nominal diameter of

3.5 in. to between 1.0 and 1.5 in.[7] Corrosion occurred in the area of the stud adjacent to the top surface of the lower flange. Visual inspection revealed that the severely corroded studs had an hour-glass appearance. Visual inspection of the closure studs at the other PWR plants has revealed that the studs in all different pump designs are susceptible to boric acid corrosion. Installation of instrumentation for actively monitoring the leak-off lines between the gaskets is necessary to detect leakage. If the leak-off line installed between the gaskets is plugged or not instrumented, no indication of reactor coolant leakage from the inner gasket will be available.

The major causes of gasket leakage are poor maintenance, minor corrosion of the stainless steel portion of the gasket, and poor gasket spring-back.[5] To improve plant reliability, the following recommendations are made to upgrade the maintenance procedures for the closure studs: use of gaskets with improved spring-back characteristics, proper procedures for gasket installation, and proper fastener tensioning practices. Proper use of fastener lubricants and injection sealants is also recommended.

Stress Corrosion Cracking. Cast stainless steel pump casing and its weldments have excellent resistance to stress corrosion cracking. However, if very low levels of ferrite are present at the welds because of the filler material and weld procedures used, both repair and fabrication weldments in pumps could be sensitized and become susceptible to environmentally induced stress corrosion cracking.[18] The full-solution heat treatment reduces or eliminates the sensitization in the weldments in Types E and C pump casings. To prevent additional sensitization, and also to reduce residual stresses in Types E and C pump weldments, all welding performed after the full-solution heat treatment of the casings is limited to a low heat input no greater than 19,700 J/cm (50,000 J/in.). Therefore, the susceptible sites in the Types E and C pump casings will be at the weldments connecting the pump casing and the reactor coolant piping, if the ferrite content is very low.

The leakage of the reactor coolant across the pump casing-to-cover gasket may wet the insulation and cause a chloride attack on the pump casing. To prevent a chloride attack, the insulation must meet the acceptance criteria for the chloride concentration as specified in Regulatory Guide 1.26. In addition, if leakage takes place, adjacent areas of the pump casing may become susceptible to stress corrosion cracking.

In-Service Inspection

Pump Casing. At least one reactor coolant pump from the reactor coolant system is generally disassembled for inspection and maintenance at the end of an inspection interval. In-service inspection requirements for the pump casing include surface and volumetric examination of repair and fabrication welds. The cast stainless steel pump casings are difficult to inspect with conventional ultrasonic testing methods because of the elastic anisotropy caused by the different grain structures in the castings, and severe attenuation of the ultrasonic wave caused by the coarse grains in the steel. Therefore, radiography is generally used for volumetric examination of cast stainless steel pump casing welds. The indication detected by radiography must be considered as a worst flaw, i.e., a surface flaw with an aspect ratio of 0.5, because the radiography cannot determine the location or size of the flaw. Triangulation radiography may be used to locate the flaw. Radiography requires access to both sides of the casing wall, which may result in high cost.

The acceptance standards for the allowable length of the surface and subsurface flaws are given in the Table IWB-3518-2 of ASME Code, Section XI; the allowable surface flaw length for pump casings is less than half the length of the subsurface flaw.[19] ASME Code, Section XI permits a surface examination (e.g., liquid penetrant test) to determine whether a detected flaw is a surface flaw and, if so, the length of the flaw. However, considerable surface smoothing of the rough casting surfaces may be necessary to make meaningful surface examination and interpretation of the results.

One of the major concerns for in-service inspection of cast stainless steel components is that the extent of thermal embrittlement may be such that an existing flaw can reach critical dimensions. Currently, advanced ultrasonic testing methods are being developed to detect a flaw and determine its size, orientation, and location, to ensure the continued structural integrity of the cast stainless steel components.[18,20] Radiography has been proven adequate to detect flaws in cast components, but it cannot characterize the flaw; and it might be an expensive alternative to ultrasonic testing (UT), both in downtime and radiation exposure.

The in-service inspection requirements were originally developed for Type F pump casings, which have high residual stresses at the welds. However, these requirements may not be practical or meaningful for Types C and E pumps having different geometric configurations and fabrication, which includes complex welds but low residual stresses.

If the reactor coolant leaks, the wet insulation may create the possibility of a chloride attack on the adjacent pump casing wall. Therefore, the affected area of the pump casing may be examined by the penetrant testing method to detect any damage from stress corrosion cracking.

Pump Shaft. The LWR coolant pump is generally equipped with two vibration pickups mounted at the top of the motor stand in a horizontal plane to pick up radial vibrations of the pump. Proximity probes have also been used for vibration monitoring to detect circumferential cracks in the pump shaft. Monitoring of pump motor frame vibrations has been successfully used to indicate damage to the pump shaft.[14,15] The inspection should include surface and volumetric examination of the pump shaft. Several utilities have used the conventional UT technique to inspect pump shafts, but the results have been inconclusive and misleading. A new UT technique, i.e., a cylindrically guided wave technique, is being developed by the Southwest Research Institute for shaft inspection, and the initial results of its use are promising.[21,22]

Pump Closure Studs. In-service inspection requires volumetric examination of all the bolts, studs, nuts, and bushings during each inspection interval. However, the conventional UT techniques are not effective in revealing corrosion wastage of the studs.[23] Therefore, in-service inspection requirements, which are currently limited to volumetric examinations, should be supplemented with visual examinations. If the insulation covering the studs is removable, removal will facilitate visual examinations. Use of the cylindrically guided wave technique developed by the Southwest Research Institute may be considered, because it is capable of detecting both flaws and corrosion wastage in the long studs.[24] Surface examination of the flange surfaces is required when a mechanical joint is disassembled.

Summary, Conclusions, and Recommendations

Problems associated with time-dependent aging, such as fatigue, radiation and thermal embrittlement, corrosion, and other effects have been occurring in U.S. light water reactors as they have matured. Therefore, effectively managing the aging-related problems in older plants has become a major focus of the research sponsored by the USNRC.

The Aging Assessment and Mitigation Project is being carried out at the Idaho National Engineering Laboratory to help the USNRC understand, detect, and mitigate the aging of the major light water reactor components. The assessment of the degradation processes active in the major components has been completed. The assessment of the degradation processes active in the PWR primary coolant pumps and BWR recirculation pumps is presented in this paper. Table 3 summarizes the stressors, degradation sites and mechanisms, and potential failure modes for the pump casing, shaft, and closure studs. The conclusions and recommendations related to aging degradation of pump casing, closure studs, and pump shaft are as follows:

1. Thermal embrittlement is the primary degradation mechanism for the reactor coolant pump bodies made of cast austenitic-ferritic (duplex) stainless steel. Large ferrite content (18 to 22% are not uncommon) make thick pump casing more susceptible to damage from thermal embrittlement. Inasmuch as thermal embrittlement causes the slow loss of material fracture toughness, the actual degree of thermal embrittlement needs to be determined to assess the structural integrity of the pump casing. It is recommended to develop a model to estimate degradation in fracture toughness as a function of coolant temperature, time of exposure at temperature, chemical composition, and ferrite content and its spacing in micro-structure. Other potential methods to determine the degree of thermal embrittlement are ultrasonic testing techniques and mechanical testing of small specimens removed from the components.

2. Ferrite distribution through the statically cast, thick-wall, cast stainless steel components is not uniform. Therefore, data for the ferrite distribution in a pump casing are needed to determine the actual degree of thermal embrittlement.

3. The existing flaws in statically cast components are shrinkage flaws introduced during fabrication. Thermal embrittlement may be such that an existing flaw can reach critical dimensions that would lead to disruptive failure of casing. Therefore, it is essential to characterize the flaws to ensure the continued structural integrity of the pump casing. Advanced ultrasonic methods are needed for this purpose.

4. ASME Section XI, Table IWB-3518-2, provides standards for allowable planar flaws in the cast stainless steel pump casing welds. Similar standards are needed for the flaws in the base metal, and they should take into account degradation from thermal embrittlement.

5. Laboratory tests show that the loss of toughness caused by thermal embrittlement can be recovered by annealing at 550°C for one hour, followed by rapid cooling to lower temperatures. However, annealing of pump casing is feasible only with slow heating and cooling to and from the annealing temperature, which will cause formation of several other phases in ferrite resulting in additional loss of toughness. Therefore, annealing is not an acceptable solution.

6. Cast stainless pump casing and its welds have excellent resistance to stress corrosion cracking. However, if very low levels of ferrite are present at the welds because of the filler material and weld procedures used, both repair and fabrication welds become susceptible to stress corrosion cracking. As the Types E and C pump casings are subject to full-solution heat treatment, the susceptible sites will be at the weldments

TABLE 3. SUMMARY OF DEGRADATION PROCESSES FOR PWR COOLANT PUMPS

Rank	Sites Degradation	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Methods
1	Pump body (casting)[a]	Temperature, system operating transients, LWR environment	Thermal aging, fatigue, stress corrosion cracking	Through-wall leakage, unstable ductile tearing	Volumetric, surface
2	Closure studs	Leakage of borated coolant in PWR	Corrosion, wastage	Leakage, breakage	Volumetric
3	Pump shaft	Alternating bending stresses, rapidly changing thermal stresses, and residual stresses	Fatigue	Breakage (contained by pump body)	Surface, volumetric

---

a. Wrought carbon steel pump bodies make up a very small percentage of those in the field and have not had a long enough service history to be reflected here.

connecting the pump casing and the reactor coolant piping, if ferrite content is very low. It is suggested to determine ferrite content in the pump casing welds.

7. Weldments in Type F pump and high-stress regions in Types C and E pumps are susceptible to fatigue damage. In addition, the presence of microfissures in the weldments having a lower ferrite content (<3%) may adversely affect the fatigue strength of the pump casing and should be taken into account. The fatigue damage is expected to be quite small in the absence of microfissures.

8. The ASME Section XI in-service weld inspection requirements were originally developed for the Type F pumps, which have high residual stresses at the welds. However, these requirements may not be practical or meaningful for Types C and E pumps because of their different geometric configurations and fabrication, which includes complex welds but no residual stresses. The high stress region in Types C and E pumps is likely to include some portion of both the base metal and weld regions. Surface examination of the high stress regions is recommended.

9. The leakage of borated water across the pump case-to-cover gaskets has resulted in corrosion and wastage of pump closure studs. In carbon steel pump bodies, this leakage is also likely to cause corrosion of the base metal. The corrective actions to prevent leakage include the use of gaskets with better spring-back characteristics, proper gasket installation and cleanliness control, and proper stud tensioning practices. To detect leakage, the leak-off lines between the inner and outer gaskets should be left unplugged and instrumented.

10. ASME Section XI, Table IWB-2500-1, requires volumetric examination of the closure studs. However, the conventional ultrasonic techniques are not effective in revealing corrosion wastage of the studs. Therefore, current in-service inspection requirements for the closure studs should be supplemented with visual examinations. The use of the cylindrically guided wave technique developed by the Southwest Research Institute may be considered because it is capable of detecting both cracks and corrosion wastage in the long studs.

11. Use of conventional ultrasonic techniques to inspect pump shaft gives inconclusive and misleading results. Field application of cylindrically guided wave technique for shaft inspection needs to be evaluated. Monitoring of the pump motor frame vibrations is recommended to detect the damage to the pump shaft.

References

1. J. P. Vora, Nuclear Plant Aging Research (NPAR) Program Plan, Rev. 1, NUREG-1144, September 1987.

2. V. N. Shah and P. E. MacDonald (eds.), Residual Life Assessment of Major Light Water Reactor Components - Overview, 1, NUREG/CR-4731, EGG-2469, June 1987.

3. V. N. Shah and P. E. MacDonald (eds.), Residual Life Assessment of Major Light Water Reactor Components - Overview, 2, NUREG/CR-4731, EGG-2469 (Draft), March 1988.

4. V. N. Shah and P. E. MacDonald, "Residual Life Assessment of Major PWR Components," Proceedings of the International Topical Meeting on Operability of Nuclear Power Systems in Normal and Adverse Environments, Albuquerque, New Mexico, September 29-October 3, 1986, pp. 112-119.

5. F. R. Drahos, W. L. Server, and B. F. Beaudoin, "Light Water Reactor Coolant Pumps," in Residual Life Assessment of Major Light Water Reactor Components - Overview, 2, V. N. Shah and P. E. MacDonald (eds.), NUREG/CR-4731, EGG-2469 (Draft), March 1988, pp. 8-29.

6. D. T. Umino and A. K. Rao, Long-Term Inspection Requirements for PWR Pump Casings, EPRI NP-3491, May 1984.

7. "Degradation of Reactor Coolant Pump Studs," U.S. Nuclear Regulatory Commission IE Information Notice No. 80-27, June 1980.

8. Personal communication with Dr. Kassner, ANL, May 1988.

9. U.S. Nuclear Regulatory Commission, "Control of Ferrite Content in Stainless Steel Weld Metal," Regulatory Guide 1.31, Revision 3, April 1978.

10. Personal communication with E. Landerman, Westinghouse, September 1988.

11. O. K. Chopra and H. M. Chung, "Effect of Low-Temperature Aging on the Mechanical Properties of Cast Stainless Steels," in Properties of Stainless Steel in Elevated Temperature Service, M. Prager (ed.), ASME, MPC-Vol. 26, PVP-Vol. 132, December 1987, pp. 79-112.

12. Personal Communication with Dr. W. J. Shack, Argonne National Laboratory, August 1988.

13. C. E. Jaske, Quantification of the Degradation of Cast Stainless-Steel Components Caused by Thermal Embrittlement, Report being prepared for EG&G Idaho.

14. S. M. Stoller Corporation, "RCP Shaft Fractured, Cap Screws Cracked, Broken - Fabrication, Thermal and Weld Stresses, IGSCC, Insufficient Preload," Nuclear Plant Experience, PWR-2, p. 52.

15. S. D. Leshnoff and P. C. Riccardella, "Reactor Coolant Pump Shaft Crack Investigations at TMI-1," presented at the EPRI Reactor Coolant Pump/Recirculation Pump Monitoring Workshop, EPRI Component Monitoring and Diagnosis Technology Transfer Center, Toronto, Canada, March 29, 1988.

16. E. Hiruo, Inside NRC, No. 25, December 7, 1987, p. 8.

17. S. M. Stoller Corporation, "RCP Gasket Leaked - Closure Studs Corroded," Nuclear Plant Experience, PWR-2, p. 34.9.

18. G. R. Egan et al., Inspection of Centrifugally Cast Stainless Steel Components in PWRs, EPRI NP-5131, June 1987.

19. "Rules for In-Service Inspection of Nuclear Power Plant Components," 1986 ASME Boiler and Pressure Vessel Code, Section XI.

20. P. Jeong and F. Ammirato, "Nondestructive Examination of Coolant Pump Welds - Ultrasonic Examination of Cast Stainless Steel Components," presented at the Pressure Vessel and Piping Conference, Pittsburgh, Pennsylvania, June 19-23, 1988.

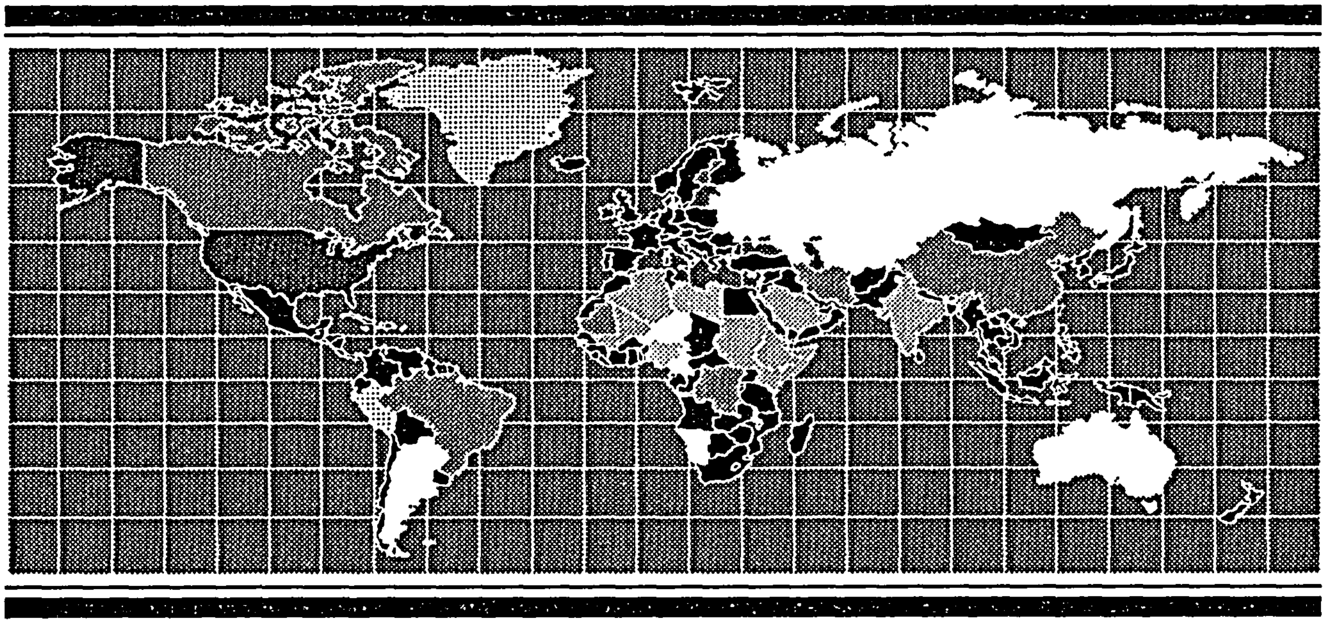
21. Electric Power Research Institute, "Cylindrically Guided Wave Technique (CGWT) for Pump Shaft Inspections," Technical Brief, 1987.

22. G. M. Light, E. A. Bloom, and S.-N. Liu, "Application of the Cylindrically Guided Wave Technique," Presented at the Pressure Vessel and Piping Conference, Pittsburgh, Pennsylvania, June 19-23, 1988.

23. U.S. Nuclear Regulatory Commission, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," Generic Letter 88-05, March 1988.

24. G. M. Light, N. R. Joshi, and S.-N. Liu, "Stud Bolt Inspection Using Ultrasonic Cylindrically Guided Wave Technique," Improved Technology for Critical Bolting Applications, E. A. Merrick and M. Prager (eds.), ASME, New York, July 1986, pp. 31-38.

Notice



TECHNICAL SESSION 5

Reliability

August 31, 1988

Session Chairman

DR. BRIAN W. SHERON
Director, Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Session Co-Chairman

ROBERT BAER
Chief, Engineering Issues Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

INTEGRATION OF ENGINEERING INFORMATION AND RISK INFORMATION FOR AGING ASSESSMENTS

W.E. Vesely
Science Applications International Corporation
J.P. Vora
U.S. Nuclear Regulatory Commission

ABSTRACT

Presently a gap exists between hardware oriented engineering evaluations of aging impacts and risk evaluations of aging impacts. The engineering evaluations cannot be readily related to their reliability and risk impacts, and the risk evaluations cannot directly utilize engineering information. A framework is presented in this paper for integrating the engineering evaluations and the risk evaluations of aging impacts. Specific relationships which need to be developed for the engineering-risk integration are defined and methods for obtaining the relationships are discussed. The impact these relationships can have in understanding and in managing aging is described. Finally, a program is presented which can be followed in developing the engineering-risk relationships for specific components and structures.

1.0 INTRODUCTION

Assessments of aging impacts on component and structure performance involve engineering evaluations and can involve risk evaluations. Engineering evaluations of aging impacts consist of evaluations of the changes in the physical properties of the component or structure versus its age. The associated aging mechanisms and failure modes of the component or structure are also identified.

Examples of engineering evaluations of aging impacts which have been carried out as part of NRC's Nuclear Plant Aging Research (NPAR) Program include evaluations of electric motor performance,[1] evaluations of the aging and service wear of auxiliary feedwater pumps,[2] evaluations of the aging of diesel generators,[3] evaluations of the aging and service wear of electric motor operated valves,[4] and evaluations of concrete aging.[5] In addition, studies have been performed on batteries, chargers, circuit breakers, relays, check valves, and solenoid valves. Also aging-system interaction studies have been performed on six different electrical and fluid-mechanical systems. These engineering evaluations for the most part are descriptive in nature, describing the failure mechanisms, physical property changes, and failure modes which have been observed for aged components or structures. Current inspection, surveillance, and maintenance practices are also described in light of this information.

Risk evaluations of aging impacts consist of evaluations of the system unavailability impacts and the risk impacts from aging components. The risk evaluations of aging impacts use as input component failure rates versus age, which describe in a reliability sense the aging of the components. With the given component failure rates versus age, reliability and risk formulas are used to calculate the system unavailability and plant risk versus age.

An example of risk evaluations of aging is the evaluation using the linear aging model to describe the component failure rate versus age.[6] The linear aging model describes the failure rate of an aging component as a linear function of the component's age. Other models are also available which can be used to define the failure rate of an aging component.[7,8] The unknown parameters in these failure rate models are generally estimated from failure times of the component using various statistical techniques. If data do not exist then the failure rates are subjectively estimated. The failure rates versus age for the different components are then input into age dependent risk models to calculate the age dependent system unavailabilities and risk results.[9,10]

With the present state of the art of risk evaluations of aging impacts and with the present state of the art of engineering evaluations of aging impacts, there is no interface between these two evaluations. This gap between engineering evaluations and risk evaluations causes deficiencies in each of the evaluations. The engineering evaluations are not able to explicitly determine the reliability or risk impacts of the engineering information which is obtained. The risk evaluations in turn are not able to relate the component failure rates to detailed engineering information. Instead the failure rates are based on observations of failures, which generally are rare, or are based on expert judgment, which has large associated uncertainties.

The subsequent sections in this paper present a framework for integrating engineering evaluations of aging impacts with risk evaluations of aging impacts. Requirements are presented as to what engineering information and what risk information needs to be supplied to integrate the two evaluations. With this integration, the risk and reliability impacts of engineering evaluations can be determined, and the aging component failure rates can be explicitly related to engineering information. Examples are given to illustrate the requirements, and implementations are discussed. A program is presented which can be carried out to develop this integration for specific components and specific structures.

2.0 THE PHYSICAL PROPERTY RELATIONSHIPS WHICH NEED TO BE DETERMINED

To be able to integrate engineering evaluations of aging impacts and risk evaluations of aging impacts, the first type of information which is needed is explicit physical property versus age relationships. These relationships describe how the physical properties of the component or structure change with age under given environments. The physical property relationships which need to be determined can be symbolically represented as:

$$p = f(t,E) \qquad (1)$$

where p is a physical property value (or change in physical property value) at age t under environmental conditions E. The specific form of the function f(t,E) needs to be determined.

The above relationship needs to be determined for each of the physical properties of the component or structure which are assessed to be important to the functioning of the component or structure. In these relationships, t is an appropriate measure of the age of the component or structure, and E is the set of environmental variables which affect the property change with age.

The relationships f(t,E) which are determined can be crisp, mathematical relationships, which are the most definitive relationships. Alternatively, the relationships which are determined can be fuzzy relationships which are less definitive, but still are usable. An example of a crisp, mathematical relationship is the linear model

$$p = at + b \qquad (2)$$

where a and b are parameters which give the deterioration of p with the age. For example, p may be a wall thickness, and the values of a and b give the deterioration of the wall thickness versus age. Alternatively, p may be the degree of wear of a pump rotor which is measured in an appropriate dimension. For a dielectric insulating material p could be a dissipation factor or a loss factor subject to age deterioration. For such models, the associated uncertainties will also need to be determined and be quantified.

An example of a fuzzy relationship which can be determined for f(t,E) is the following descriptive model:

> "Property p undergoes insignificant changes if the age is less then t_1, undergoes moderate changes if the age is between t_1 and t_2, and undergoes severe changes if the age is greater than t_2.

The fuzzy terms "insignificant change," "moderate change," and "severe change" imply a range of possible property values. The ages t_1 and t_2 can also be fuzzy. The disciplines of fuzzy set theory[11,12] and Bayesian analyses[13,14] can be used to translate the above descriptive-type relationships into quantitative relationships.

The above required physical property relationships generally have not been determined in the engineering studies which have been carried out to date in NPAR. However, in a number of these studies, sufficient information has been collected so that these relationships could be developed. Figure 1, for example, shows the relationship of concrete tensile strength versus neutron fluence which is identified in Reference 5. By relating the neutron fluence versus age, the concrete tensile strength versus age could then be developed. Other similar relationships, most of which are fuzzy in nature, can also be obtained from the NPAR work.

3.0 THE MAINTENANCE RELATIONSHIPS WHICH NEED TO BE DETERMINED

The previous physical property relationships f(t,E) can incorporate the effects of present maintenance practices. However to specifically evaluate maintenance strategies, separate maintenance relationships need also to be determined. This is the second type of information that needs to be determined.

The maintenance relationships quantify the explicit effect of maintenance on the physical property changes which are caused by aging. The maintenance relationships can again either be definitive or fuzzy, but they need to be explicit. An example of a definitive relationship is:

> "The effect of maintenance is to improve the property value to P_o.

For example, if p is the amount (depth) of erosion on component piecepart then p_o will be the value after the maintenance. For component replacements or overhauls, p_o would be the original value of the physical property. Other definitive descriptions of maintenance effects could quantify the change in physical property produced by the maintenance. In these relationships, the associated uncertainties would again need to be quantified as part of the definitive maintenance descriptions.

An example of a fuzzy evaluation of the effects of maintenance is:

> "The effect of maintenance is to moderately improve the property value."

Fuzzy set theory approaches and Bayesian statistical techniques can again be used to translate the fuzzy description into a corresponding quantitative description.

A number of the engineering studies which have been carried out in NPAR have evaluated the effectiveness of present maintenance activities in controlling or arresting aging effects on specific component properties. For example, Reference 1 evaluates the effectiveness of maintenance activities in controlling the wear of auxiliary feedwater pumps. As taken from Reference 1,

> "The only present maintenance activity pertains to shaft sealing components. ...With stuff-box shaft seals, the primary maintenance activity is to keep the packing gland properly sealed,..."

Recommendations are also made:

> "The disassembled coupling should be inspected for any deterioration of the gear teeth. If the coupling shows measurable wear or tooth surface deterioration then the coupling should be replaced."

It would not be difficult to translate these maintenance evaluations and recommendations and other similar ones in the NPAR work to the required maintenance relationships for the present maintenance activities as well as for the recommended maintenances.

4.0 THE FAILURE RATE RELATIONSHIPS WHICH NEED TO BE DETERMINED

The third and last type of information which needs to be determined to integrate engineering evaluations and risk evaluations is the identification of explicit relationships which relate component failure rate changes to physical property changes. The failure rate relationships which need to be determined can be symbolically expressed as:

$$\lambda = g(p) \qquad (3)$$

where λ is the component or structure failure rate and p is the set of important physical properties of the component or structure. Equivalently, λ can be the failure rate change and p the physical property change. The specific form of the function g(p) needs to be explicitly determined.

The function g(p) needs to be determined for each of the critical failure modes of the component or structure. If environmental variables affect the failure rate (apart from the physical properties) then the environmental variables will appear as additional variables in the relationship.

The relationship g(p) can again either be a definitive, mathematical relationship or can be a fuzzy relationship. Because of the vagueness of available information, the relationship will often be fuzzy.

An example of a definitive, mathematical relationship is the linear failure rate model,

$$\lambda = cp + d \qquad (4)$$

where the parameters c and d define the change in the failure rate with the physical properties. The parameters with their uncertainties could be estimated using various statistical and solicitation approaches.[15,16,17]

Alternatively, the failure rate relationship which is determined can be a fuzzy relationship, such as,

> "For a physical property value less than p_1 the failure rate change is insignificant, for values between p_1 and p_2 the failure rate change is moderate, and for physical property values greater than p_2, the failure rate change is significant."

The property values p_1 and p_2 can also be fuzzy. Fuzzy set theory approaches and Bayesian approaches can again be used to translate the above descriptive-type models into quantitative models.

The engineering studies performed in NPAR again did not determine explicit relationships for $\lambda(p)$, however the information which was obtained in a number of these studies could be used to determine relationships for $\lambda(p)$. For example, Reference 3 categorizes diesel failure mechanisms and fracture causes, which include the resulting physical property changes, according to their component failure potential. Reference 2 categorizes feedwater pump failure mechanisms and causes, and their resulting physical property changes, according to their relative probability of occurrence. These results could be used in a straightforward manner to determine fuzzy relationships for $\lambda(p)$.

5.0 IMPLICATIONS OF THE INTEGRATION RELATIONSHIPS

Having determined the physical property relationship and the failure rate relationship, we then have:

$$p = f(t,E) \qquad (5)$$

and

$$\lambda = g(p) \qquad (6)$$

The component or structure failure rate is related to its physical properties by Equation (6), and the physical properties are related to its age by Equation (5). Also, having determined the maintenance relationship we can quantify the effects of maintenance on the physical properties and hence on the failure rate. Figure 2 illustrates the age-physical property-failure rate relational space for linear relationships for p and λ given by Equations (2) and (4).

We have the following implications and implementations from these relationships, once they have been developed:

1. The component or structure failure rate has a physical property basis and is related to its age through its physical properties. This is in contrast to present failure rate formulas which have empirical constants that must be estimated from failure occurrences or from subjective estimates of the failure rate.

2. The reliability and risk implications of observed physical properties can be determined by using the associated failure rate value in reliability and risk models. The research and engineering investigations which have been carried out can thus be translated into their reliability and risk implications.

3. Maintenance can be evaluated for its effectiveness in controlling not only physical property deteriorations but also in controlling risks. The risk effectiveness of maintenance can be determined by translating changes in physical properties produced by maintenance into the corresponding failure rate changes and then evaluating the risk effects.

and

4. Condition monitoring and maintenance monitoring can be carried out on a physical property level, as well as on a failure and unavailability level. Alarm limits can be placed on the physical properties to alarm when the failure rate or risk is becoming unacceptable, before actual failures or incidents occur.

6.0 TASKS FOR A RESEARCH PROGRAM

From the preceding section, it is apparent that there are significant benefits to be gained from developing the relationships which integrate engineering evaluations of aging impacts with risk evaluations of aging impacts. Table 1 lists the tasks which can be carried out to develop the integrating relationships for specific components and structures. The relationships which are developed likely will be quite imprecise and will be quite fuzzy. However, reliability and risk analyses do not require precise models since order of magnitude precision (a factor of 10 error) is generally acceptable. As experience is gained, the degree of impreciseness and fuzziness will also likely decrease. To deal with the expected fuzziness, the components or structures which are first selected can be those for which there is the greatest amount of information available on property changes versus age and on failure behaviors.

7.0 SUMMARY

The first step needed to integrate engineering and evaluations of aging is the development of relationships which explicitly describe how physical properties change with age. These physical property relationships can incorporate the effects of present maintenance practices, however to explicitly evaluate the effects of maintenance, separate maintenance relationships need also to be developed. These maintenance relationships describe the explicit effects that maintenance activities have on the physical properties.

To relate the physical property changes to failure rate changes, the final step in the integration is to develop the failure rate relationships which explicitly describe how the failure rate changes with the physical property changes. The explicit failure rate relationships will establish a physical property basis for the failure rate, resulting in engineering-based risk assessments of aging. The maintenance relationships will furthermore provide an engineering basis and risk basis for maintenance effectiveness evaluations.

Utilizing the relationships will allow the risk effects of physical property observations to be determined. Condition monitoring and performance monitoring of physical properties can be related to risk and reliability effects, thereby greatly improving the effectiveness and potentials of these processes. Tasks are presented which can be carried out to develop these relationships and to realize these potential benefits for specific components and structures.

TABLE 1. TASKS TO DEVELOP ENGINEERING-RISK RELATIONSHIPS

1. Select the components or structures for which the engineering-risk relationships are to be developed.
2. Identify the important physical properties which affect the reliability and the residual life (or time of failure).
3. Collect information on the change of the physical properties versus age, where age is measured in relevant units.
4. Develop the explicit physical property relationships which describe the change in physical properties versus age; use statistical approaches and fuzzy set approaches to describe the associated uncertainties.
5. Determine the nominal failure rate of the component or structure when no aging exists; this will serve as the reference failure rate.
6. Develop the explicit failure rate relationship which describes the change in failure rate versus physical property change; use statistical approaches and fuzzy set approaches to describe the associated uncertainties.
7. Review the physical property and failure rate relationships for their validity; review also the implications of the relationships.
8. Develop the explicit maintenance relationships which describe the effect of specific maintenance activities on the physical properties which have deteriorated with age.
9. Review the maintenance relationships for their validity: review also the implications of the relationship.
10. Utilize the physical property, failure rate, and maintenance relationships in time dependent and age dependent risk models to evaluate the risk implications of aging and to study age control strategies.

REFERENCES

1. M. Subdhi et al., "Improving Motor Reliability in Nuclear Power Plants," NUREG/CR-4939, November 1987.

2. M.L. Adams and E. Mackay, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Power Plants," NUREG/CR-4597, July 1986.

3. K.R. Hoopingarner et al., "Aging of Nuclear Station Diesel Generators: Evaluation of Operating and Expert Experience," NUREG/CR-4590, August 1987.

4. D.M. Eissenberg et al., "Aging and Service Wear of Electric Motor-Operated Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants," Oak Ridge Technical Report, March 22, 1985.

5. D.J. Naus, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants," NUREG/CR-4652, September 1986.

6. W.E. Vesely, "Risk Evaluations of Aging Phenomena: The Linear Aging Model and Its Extensions," NUREG/CR-4769, April 1987.

7. B.V. Gnedenko et al., Mathematical Methods of Reliability Theory, Academic Press, New York, 1986.

8. N.R. Mann, R.E. Schafer, and N.D. Singpurwalla, Methods for Statistical Analysis of Reliability and Life Data, Wiley and Sons, New york, 1974.

9. W.E. Vesely and F.T. Goldberg, "FRANTIC-A Computer code for Time Dependent Unavailability Analysis," NUREG-0193, 1977.

10. T. Ginzburg and J.T. Powers, "FRANTIC III - A Computer Code for Time Dependent Reliability Analysis," Brookhaven National Laboratory Technical Report A-3230, 1986.

11. A. Kandel, Fuzzy Mathematical Techniques with Applications, Addision-Wesley, Reading, Massachusetts, 1986.

12. G.J. Klir and T.A. Folger, Fuzzy Sets, Uncertainty, and Information, Prentice Hall, Englewood Cliffs, New Jersey, 1988.

13. G.E. P. Box and G.C. Tiao, Bayesian Inference in Statistical Analysis, Addison-Wesley, Reading, Massachusetts, 1973.

14. F. Mosteller and D.C. Wallace, Applied Bayesian and Classical Inference, Springer Verlag, New York, 1984.

15. J.O. Berger, Statistical Decision Theory and Bayesian Analysis, Springer Verlag, New York, 1985.

16. R.M. Hogarth, "Cognitive Processes and the Assessment of Subjective Probability Distributions," Journal of the American Statistical Association, 70, pp. 274-289, June 1975.

17. N.D. Singpurwalla, "An Interactive PC-Based Procedure for Reliability Assessment Incorporating Expert Opinion and Survival Data," Journal of the American Statistical Association, 83, pp. 43-51, March 1988.

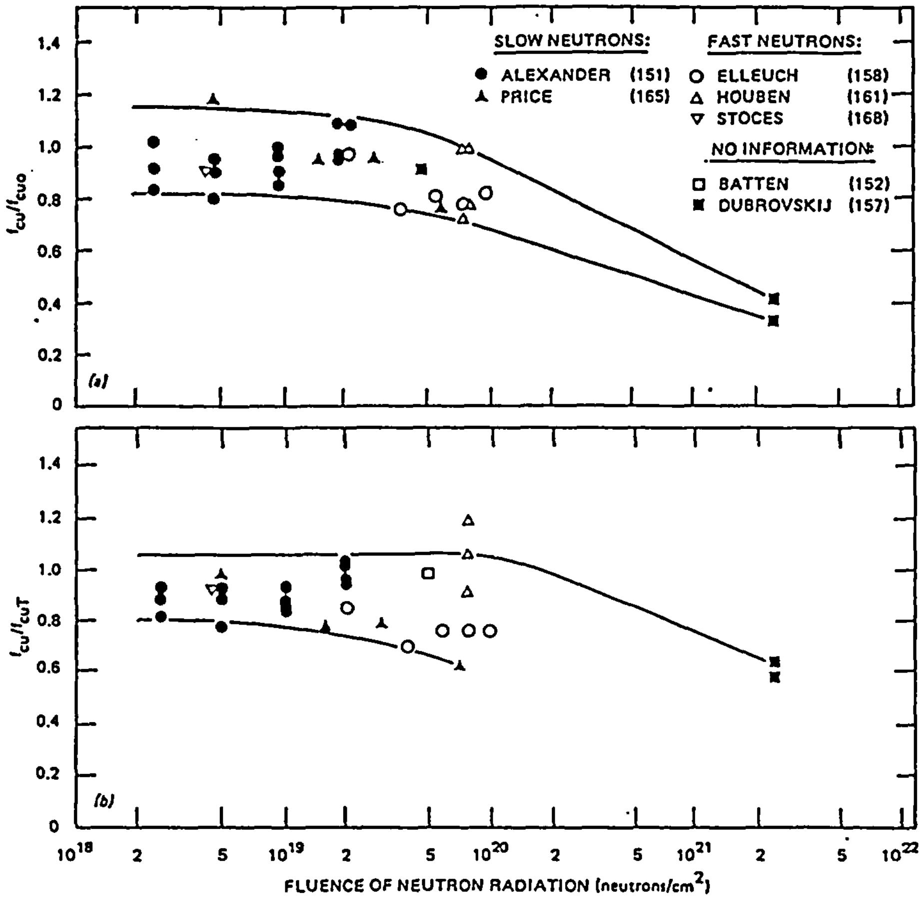


FIGURE 1. RELATIVE COMPRESSIVE STRENGTH OF CONCRETE VERSUS AGE (TOP GRAPH-THERMAL EFFECTS NOT INCLUDED, BOTTOM GRAPH-INCLUDED) ·

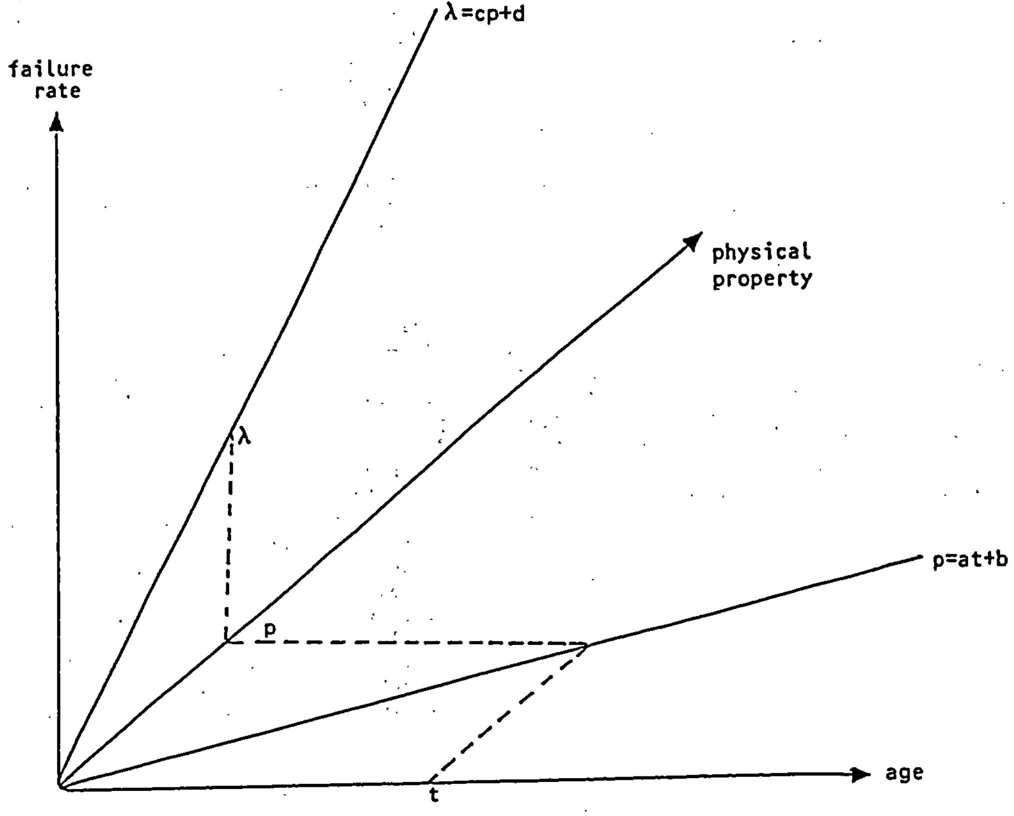


FIGURE 2. THE AGE-PHYSICAL PROPERTY-FAILURE RATE RELATIONSHIP SPACE

FAILURE DATA ANALYSIS INCLUDING AGING EFFECTS

R.W. Hockenbury,[1] R.F. Kirchner,[1] J.K. Rothert,[2] and R.J. Schmidt[2]

Summary

An extensive search and evaluation of new methods for the estimation of failure rates, including aging effects, has been performed. The immediate application is to cost-benefit calculations for the desirability of spare components in a nuclear power plant. This work emphasizes cases where few or no failures have occurred. The implications of aging effects to time-dependent failure rates are demonstrated and discussed. Potential methods for estimation of time-dependent failure rates, including censoring effects, are discussed with example applications. These methods do not seem to have been fully utilized in past safety and reliability analyses.

Introduction

Certain large, costly components in nuclear power plants, such as pumps and motors, are characterized by high reliability. However, if they do fail, in a catastrophic mode, replacements may not be readily available. Ordering, construction and shipping times may be involved thus causing extensive plant downtime. Therefore, cost-benefit studies must be performed to analyze the desirability of having spare components stored on-site. The principal quantities of interest are component reliability and availability. Thus, component failure rates and per demand failure probabilities are basic to the entire analysis. Adding new complexity to the analysis is the recent verification of aging effects[1] on some plant components.

A cost-benefit study could be divided into five parts:

1. Selection of systems to be studied.
2. Choice of probabilistic measures, i.e., change in reliability with a spare available.
3. Collection/creation of a failure data base including aging effects if appropriate.
4. Calculation of reliability measures for each system and the associated economic calculations.
5. Documentation of the overall methodology in a format which will ensure its long term use by staff engineers. Here long term use includes periodic updating of failure data and failure rate estimates, confidence limits and other statistical measures to be described below.

The main theme of this paper concerns item 3, the failure data base. Systems were selected on the basis of plant experience. Various probabilistic measures were considered. The key quantity now in use is the change in reliability vs. time. That is, the decrease in reliability in a given year reflects the probability of needing a spare for the component of interest. Our work to date has concentrated on items 1 through 4. The problems resulting from the scarcity of failure data have been cited since the beginning of risk assessment of nuclear power plants. Indeed, this circumstance has led to major efforts in data collection and analytical techniques. Some of these have been noted, for example, in the Reactor Safety study (WASH-1400) and in the PRA Procedures Guide (NUREG/CR-2300).

A literature search has been carried out to identify the most recent techniques for the estimation of failure probabilities using actual plant data. As expected, much work has been published since an earlier study (by RWH) for NEL-PIA (American Nuclear Insurers) in 1977. This includes the use of censored data, estimation of confidence limits on component reliability and cumulative failure probability and time-dependent failure rates due to aging. In this paper, techniques now in common use will be very briefly presented. Following this, some of the more promising and more recent approaches will be discussed. Our results based on some of these newer techniques will then be presented. Note that it is our intention in this work to concentrate on identifying and applying the most recent and appropriate, practical statistical methods as opposed to deriving new statistical methods, in the academic sense. The applications here and those in the future will, we expect, help to guide further efforts in the development of statistical estimation techniques. As "users" of data ourselves, we also believe that these recent methods have tended to be isolated in statistical journals, and not readily available to those who have need for them.

Selection of Systems and Large Components

The components of interest include large motors and pumps, for example. In particular, the first group under study consists of motors for the circulating water pumps, containment air recirculation fans, condensate pumps, high pressure safety injection pumps, to name a few. The approach here is to consider one subsystem at a time. Thus, we evaluate the need for a spare component relative to the reliability of the subsystem. As an example, a condensate system may consist of two trains with both required for full power plant operation. A spare motor on-site would minimize the unavailability in the event of one failure.

Choice of Probabilistic Measures

Various measures of reliability and the change in reliability were considered. There may not be a unique choice since the measure selected must be appropriate to the utility's cost-benefit calculations. These include interest charges, storage charges, present worth factors, cost of replacement power and other policy items perhaps subject to state public utility commission regulations. The measure chosen is the incremental probability, P_{US}, of using a spare component (motor) in any given year where P_{US} is the change in the motor system (i.e., condensate) reliability from year (n-1) to year (n) or:

---

[1] Rensselaer Polytechnic Institute, Troy, NY 12180
[2] Northeast Utilities Service Co., P.O. Box 270, Hartford, CT 06101

$$P_{US} = R(t_{n-1}) - R(t_n) \quad (1)$$

The exact manner in which P_{US} is used in the cost-benefit calculations will not be described since these are outside the primary focus of this paper.

Failure Data Analysis Methods in Use at Present

A comprehensive description of present statistical, classical and Bayesian, techniques is not attempted here. However, one could list general categories:

1. Classical in the sense of purely statistical, i.e., number of failures divided by the time or the number of failures over the number of demands. Associated with these are confidence limits and upper and lower bounds, for example. These are described in typical texts on statistical methods.

2. Bayesian methods have become quite prominent in nuclear power plant risk assessment because of the scarcity of plant-specific data. Representative references include the work by Apostolakis,[2] Kaplan,[3] Martz,[4] Vesely,[5] and Heising.[6] Bayesian methods have included single stage, two-stage, conjugate distributions and non-conjugate distributions, or analytical and discrete distribution approaches. Descriptions of these methods and references to the valuable work by other authors can be found in the articles cited above. Bayesian methods have been used to provide plant-specific information based on generic data and also to predict failure rates, in the absence of data, based on experience, assumed prior and likelihood distributions.

3. Graphical techniques (see for example, Ref. 7 and 8) exist for comparing observed times-to-failure to postulated probability density functions. Probability paper is used for efficiency of analysis. A significant number of data points are required for these methods thus the graphical approach is not usually appropriate for large components in the nuclear industry. For larger numbers of small components, these methods are quite valuable.

Failure Data Analysis Methods Selected and Under Consideration

Our literature search for failure rate estimation methods was undertaken in the belief that it was now time to re-assess the techniques available. Those methods now in wide use by the nuclear industry have undergone important and significant growth but have tended to follow a well-developed pattern, partly due to the need for standardization and also perhaps to the gap between the statistician and the engineer.

It is worth noting again that the search was and is being made from a user's perspective, rather than that of a statistician whose objective may be the pure development of new methodologies. Our applications may be beneficial in the latter respect, in addition to the immediate goals at hand.

The results from our ongoing search can be loosely classified according to several topics:

1. The confirmation and quantification of age-dependent failure modes and thus time-dependent failure rates.[1]
2. The increased use of censored data.[8,9]
3. The inclusion of competing risks (simultaneous failure modes).[10,11]
4. Improved Bayesian.[12]
5. Prediction intervals.[13]

The categories above actually overlap as does the applicability of the references cited in each case. In all cases, it appears that the methods have either (1) not been used in the nuclear industry or (2) have just been introduced.[1]

Also, some of the references provide auxiliary information beyond failure rates which add to the utility's knowledge about the reliability of a system or component. For example, the use of prediction intervals allows one to estimate, within a certain probability, a time interval in which the next failure may occur, based on past history.

Aging and Time-Dependent Failure Rates

Vesely[1] has examined and evaluated several extensive assessments of failure data aimed at identifying aging-induced failures in nuclear power plant systems. The data sources were primarily the Nuclear Plant Reliability Data System (NPRDS), Licensee Event Reports (LER's), Nuclear Power Experience (NPE), and the In-Plant Reliability Data System (IPRDS). Here, aging is defined to include such effects as corrosion, fatigue, wear and radiation damage. Degrading effects which are normally remedied during periodic maintenance are not included in the aging category. The fraction of failures due to aging for important components in several systems was obtained. This leads to Vesely's Linear Reliability Model. This model provides a time-dependent failure rate ($\lambda(t) = at$) where t is the time and the coefficient (a) is the aging rate derived from the fraction of aging failures (f_A), the average component exposure time (T) and the constant portion, (λ_o), of the total failure rate (λ_T). The constant term, λ_o, arises from those failure modes other than aging and corresponds to a random process.

Thus the overall failure rate λ_T is:

$$\lambda_T = \lambda_o + at \quad (2)$$

Vesely found that f_A and thus the aging rate (a) differ from one system to another, for reasons related to the conditions, environment, and stresses present in each case.

Where this linear aging model is relevant, its effects on reliability and availability must be examined. This has been done for illustrative purposes in our work using approximate data. The results shown below are not related to any specific situations at any utility. The numerical input values were arbitrarily selected, postulating that they could be regarded as reasonably representative. In Vesely's work, the value of f_A covered a very wide range.

Several illustrations of an age-dependent failure rate are presented in the RESULTS section below.

Use of Censored Data

Censored data, in particular, multiply censored data, are defined [9] as "incomplete data consisting of times-to-failure on failed units and differing running times on unfailed units...." In effect, the running times of the unfailed units play a role in the analysis depending on the particular method invoked. This has been demonstrated in Ref. 8 and also in Ref. 9. Briefly in Ref. 7, the unfailed times determine the plotting positions of the actual times-to-failure and thus the adequacy of the observed data relative to an assumed probability density function. In an application more relevant to the present paper, Nelson[9] makes use of both unfailed (censored) times and actual times-to-failure to estimate the scale parameter α for a Weibull distribution. The emphasis in this reference is on estimates and confidence limits for Weibull percentiles and reliabilities in cases where there are few or no failures. It is necessary to assume a value for the shape parameter β from other information. Sensitivity tests can be then carried out for the assumed value of β. For age-dependent failure rates, the linear model of Vesely, it is easy to show that the time-dependent part leads to a value of $\beta = 2$. Since the product of the aging rate and time, at, is a hazard function h(t), the relation between h(t) and its probability density function, f(t) is:

$$f(t) = h(t)\exp[-\int h(t')dt'] \tag{3}$$

$$f(t) = at\exp[-at^2/2] \tag{4}$$

Nelson, in effect, writes the hazard function h(t):

$$\text{where } h(t) = \frac{\beta}{\alpha}\left(\frac{t}{\alpha}\right)^{\beta-1} \tag{5}$$

$$\text{and thus: } f(t) = \frac{\beta}{\alpha}\left(\frac{t}{\alpha}\right)^{\beta-1}\exp[-\left(\frac{t}{\alpha}\right)^{\beta}] \tag{6}$$

Thus $\beta = 2$ and the scale parameter α is:

$$\alpha = \sqrt{\frac{2}{a}} \tag{7}$$

This establishes a link between the estimate of a, the aging rate of Vesely, and the scale parameter α by Nelson [9]. Nelson uses a maximum likelihood estimate, $\hat{\alpha}$:

$$\hat{\alpha} = \left[\frac{\sum_i^n T_i^{\beta}}{r}\right]^{\frac{1}{\beta}} \tag{8}$$

where the times T_i include both failed and unfailed times and r is the number of failures. He then expresses a lower C% confidence limit on α as:

$$\underline{\alpha} = \hat{\alpha}\left[\frac{2r}{\chi^2(C;2r+2)}\right]^{\beta} \text{ for } r \geq 1 \text{ failures} \tag{9}$$

and

$$\underline{\alpha} = \left[\frac{2\sum_{i=1}^{n} T_i^{\beta}}{\chi^2(C;2r+2)}\right]^{\frac{1}{\beta}} \text{ for } r \geq 0 \text{ failures} \tag{10}$$

where $\chi^2(C;2r+2)$ is the C^{th} percentile of the chi-square distribution with (2r+2) degrees of freedom. From the estimate of α, Nelson obtains the maximum likelihood estimate of the population fraction $F(t_o)$ failing by age t_o:

$$F(t_o) = 1 - \exp[-(\frac{t_o}{\hat{\alpha}})^{\beta}] \tag{11}$$

and the associated reliability, $R(t_o)$:

$$R(t_o) = 1 - F(t_o) \tag{12}$$

Competing Risks

The principle of competing risks[10,11] has already been introduced here when a total failure rate λ_T was defined as:

$$\lambda_T = \lambda_o + at \tag{13}$$

In effect, the reliability is written as the product of two competing risks:

$$R(t) = e^{-\lambda_o t}e^{-\frac{at^2}{2}} \tag{14}$$

In addition, when one failure mode is of interest[7], units taken out of service due to other failure modes can be regarded as censored relative to the failure mode of interest. To our knowledge, this principle has not been taken advantage of in nuclear data analyses. We have made use of censored data in an example to be shown in the RESULTS section.

Improved Bayesian Analysis

Improved here is taken to mean both up-to-date and in a format more amenable for use by all engineers. Vaurio[12] has described a parametric robust empirical Bayes (PREB) method for estimating failure rates for plant-specific risk analyses. Advantages are claimed relative to the maximum likelihood method and previous parametric empirical Bayesian approaches. Moment matching equations, based on observed data are used to estimate the prior distribution parameters x and y in a case where the gamma prior density is:

$$p(\lambda_i;x,y) = y\frac{(\lambda_i y)^x}{\Gamma(x+1)}e^{-\lambda_i y} \tag{15}$$

and a Poisson likelihood function is:

$$P(K_i;\lambda_i,T_i) = \frac{(\lambda_i T_i)^{K_i}}{K_i!}e^{-\lambda_i T_i} \tag{16}$$

where:

T_i = observation times

λ_i = failure rates

K_i = number of failures, i=1, 2...k.

Since the Poisson and gamma are conjugate distributions, the posterior density is also a gamma distribution with a mean value:

$$\bar{\lambda}_i = \frac{x+K_i+1}{y+T_i} \tag{17}$$

and variance:

$$\text{Var}\ (\lambda_i) = \frac{x+K_i+1}{(y+T_i)^2} \tag{18}$$

The mean and variance are easily updated as more plant data are accumulated with these simple expressions resulting from the use of conjugate distribution functions. This is to be contrasted to the more complex discrete probability distribution approach required when non-conjugate distributions are used.

The questions yet to be addressed here include:

1. Is there always a significant improvement in results based on the use of non-conjugate distributions vs. those from the simpler conjugate distribution forms?
2. Will the non-conjugate, discretized approach come into widespread use in utilities, a circumstance required for effective use of plant specific data.

As of this writing, these questions have not been fully investigated. Our observations are noted here to provoke serious consideration by others in this field.

Prediction Intervals

It would be desirable to use past samples or experience to predict a time interval which will contain the results of a future sample, with some given probability. This prediction interval method[13,14] has been applied to a situation[13] where a certain number of failures have occurred giving a sample mean time-to-failure $\bar{y}$ of 400 hours. Exponentially distributed times-to-failure were assumed. A two-sided prediction interval which will bracket the next observed time-to-failure with a 100γ% probability is obtained using the F distribution with lower and upper bounds:

$$\frac{\bar{y}}{F(2n,2;(1+\gamma)/2)} = \text{lower bound}$$

and $\bar{y}\ F(2,2n;(1+\gamma)/2) =$ upper bound where $F(2n,2;(1+\gamma)/2)$ refers to the $100(1+\gamma)/2$ percentile of the F distribution with degrees of freedom 2n and 2.

This gives a two-sided 95% prediction interval to bound the next time-to-failure after seeing 5 units run to failure. The values of F are:

$F(10,2;.975)=39.4$

$F(2,10;.975)=5.46$

Thus the bounds on the next time-to-failure are 10.2 hours and 2184 hours, respectively.

Other aspects of prediction intervals, including the Weibull, lognormal and other distributions are cited in Ref. 13 and 14. We have not yet applied these to the need for spares and the associated cost-benefit studies. However, the prediction interval method is presently under serious consideration since it has the potential for enhancing the decision-making process. To include aging effects, the Weibull form (or Vesely's model) must be introduced.

Results

This section describes illustrations of some of the methods mentioned above. It is important to note that all the "data" are artificial, having been constructed from estimates and composites of raw data sources and several systems. The numerical values, however, are typical of nuclear power plants and are not considered to be extreme. At this point in our work, we have concentrated on failure rates for components in continuous duty as opposed to per demand situations. The per demand, change of state cases will be covered in the latter part of the program.

Age-Dependent Failure Rates

A set of times-to-failure and censored times for large components was formulated as shown in Table 1. One random failure and one aging failure have been "identified."

At first, both failures are lumped together, ignoring the aging nature of one. Using Nelson's[9] approach, see Eq. (8), a failure rate λ_c is obtained:

$\lambda_c = 4.34 \times 10^{-6}$ 1/hour at the 50% confidence level as seen in Table 1.

However, separating the aging failure mode from the random failure, treating the two datum points as censoring each other and using Nelson's method,[9] we obtain a total failure rate:

$$\lambda_T = \lambda_o + \frac{2t}{\alpha^2} \tag{19}$$

$$\lambda_T = 2.71 \times 10^{-6} + 4.01t \times 10^{-11} \tag{20}$$

as listed in Table 1 under the Random and Aging Failures columns, respectively. The units of time t are in hours. Note that Nelson's method leads to the same form of $\lambda_T = \lambda_o + at$ as expected and as seen in Vesely.[1] Figure 1 displays the failure rate λ_c and the total failure rate λ_T including the aging or time-dependent part $\frac{2t}{\alpha^2}$ and the constant portion λ_o. Note that in year one, the total failure rate λ_T is primarily the constant part λ_o for this case and is less than the rate λ_c. By year five, the total failure rate λ_T exceeds the rate λ_c and continues to increase linearly. The values and slopes of the two lines in Figure 1 are representative and more or less typical of results shown by Vesely.[1] For other data, the relative values of constant part λ_o and the total rate λ_T will of course change and the implications must be considered on a case by case basis.

TABLE 1

FAILURE DATA

OBSERV. NUMBER	CENSORED	TYPE OF FAILURE RANDOM	AGING	IN-SERVICE TIME (hours)
1		1		87600
2			1	153960
3	1			66360
4	1			153960
5	1			153960

WEIBULL PARAMETERS
CONSTANT FAILURE RATE
(ALL FAILURES)

BETA = 1

λ_c = HAZARD FUNCT. = 4.34×10^{-6} 1/hour

WEIBULL PARAMETERS
TIME-DEPENDENT FAILURE RATE

RANDOM FAILURES
BETA=1

λ_o= HAZARD FUNCT.=2.71×10^{-6}

AGING FAILURES
BETA=2

$\lambda(t)$=HAZARD FUNCT.=$4.01t \times 10^{-11}$

$$\lambda_T = \lambda_o + \lambda(t)$$

For the rates above, the component reliability vs. time is shown in Fig. 2 for both λ_c and λ_T. Note that the reliability differs at year one and then crosses over near year eight, for these data. The inclusion of aging results in a lower reliability at twenty years.

A two-of-three system was examined using the component data above. The system reliability resulting from assuming a rate λ_c and a total rate λ_T is shown in Fig. 3. Again there is a crossover in magnitude with the total rate λ_T giving a lower reliability after year eight. The shapes and crossover points will always differ, depending on the individual case. As Vesely has shown, in some situations, the aging portion is less important than others.

Using the probability measure P_{US}, see Eq. (1) for the same two-of-three system, we obtained the results presented in Figure 4. Here, the rate λ_c yields a high value of P_{US} than does λ_T before year five and a lower P_{US} afterwards. The probability P_{US} based on the total rate significantly exceeds the constant rate prediction over most of the time range for these data. The peak and following decrease are due to the asymptotic approach of both to a 100% cumulative failure probability as the time becomes very large.

In Fig. 5, the equivalent availability of the two-of-three system is shown for the constant failure rate λ_c and time-dependent rate including aging, λ_T. For each rate, the equivalent availability was calculated for a spare-no spare condition.

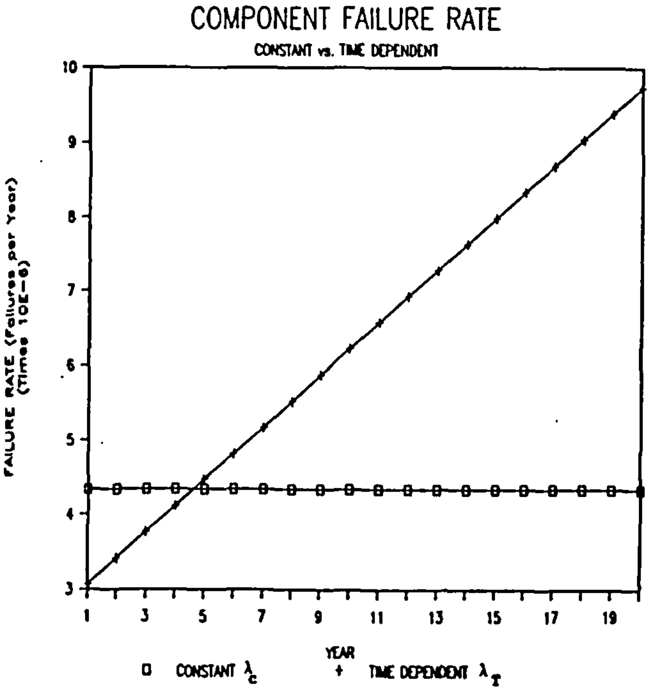


FIG. 1 FAILURE RATES λ_c AND λ_T VS. TIME.

The equivalent availability EA is defined as:

$$EA = \sum_{i=1}^{n} A_i C_i \qquad (21)$$

where: A_i = system availability in state;
C_i = system capacity in state;

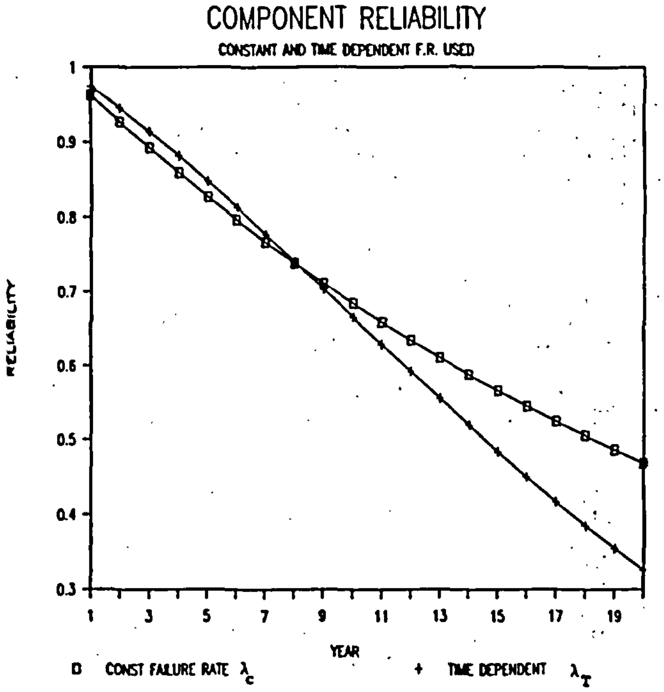


FIG. 2 COMPONENT RELIABILITY VS. TIME.

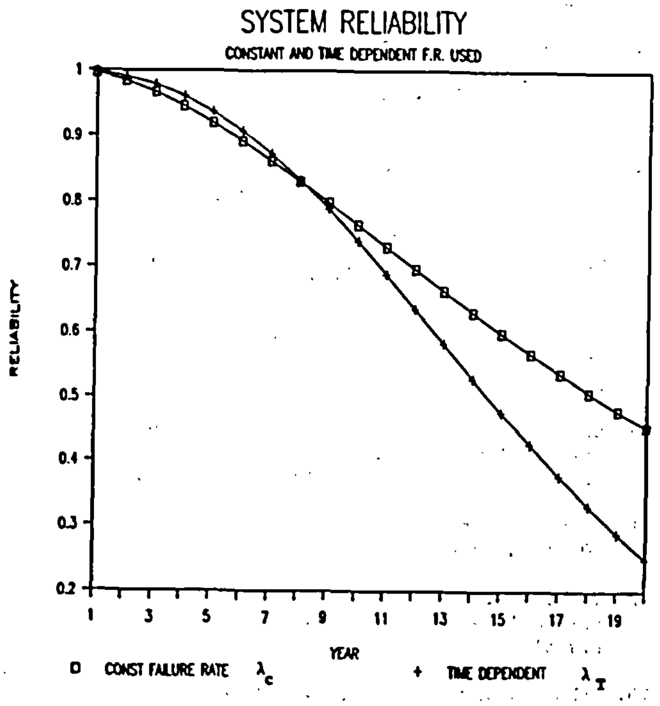


FIG. 3. TWO-OF-THREE SYSTEM RELIABILITY.

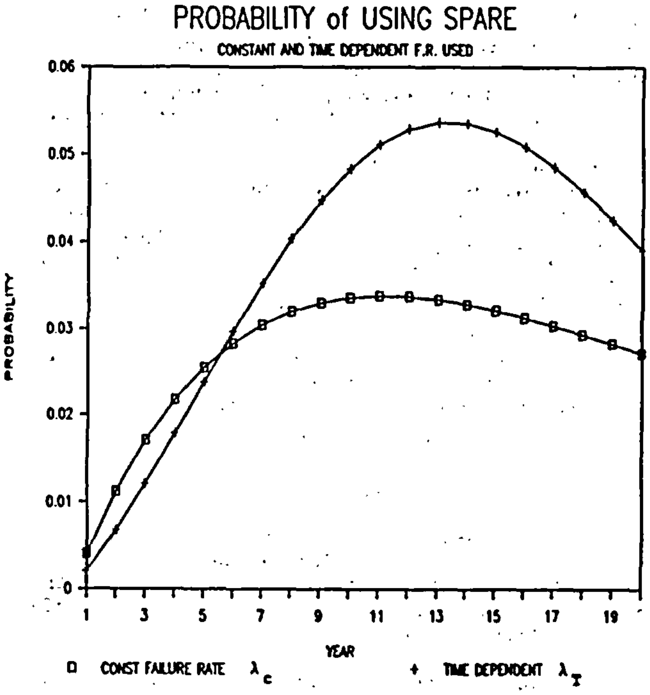


FIG. 4 PROBABILITY P_{US} OF USING A SPARE FOR 2-OF-3 SYSTEM.

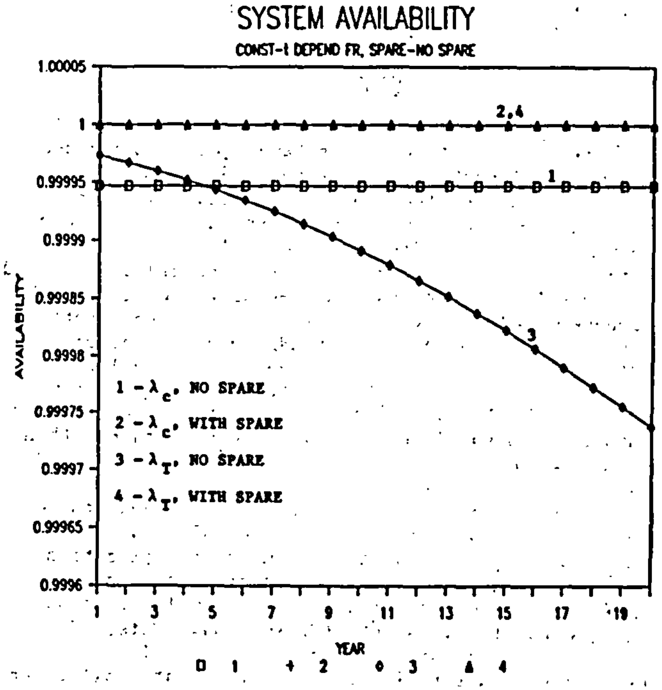


FIG 5. TWO-OF-THREE SYSTEM AVAILABILITY.

The availability A is:

$$A = \frac{MTBF}{MTBF + MTTR} \tag{22}$$

where: MTBF = Mean-time between failures.
MTTR = Mean-time to restore.

When λ_c is used, then:

$$MTBF = \frac{1}{\lambda_c} \tag{23}$$

To include aging effects, λ_T is used and:

$$MTBF\ (t) = \frac{1}{\lambda_T} \tag{24}$$

The value of MTTR depends on the spare-no-spare condition. For the "no spare" condition, we assumed that MTTR=1380 hours which included the total construction, shipping and installation time. If a spare is on-site, we assumed a 36 hour installation time.

The relative positions and shapes of the plots in Figure 5 are more significant than the absolute numbers. The availability with a spare (2,4) is better for both cases, λ_c and λ_T, relative to no spare (1,3). However, plots (using λ_c) are misleading because the true availability when aging effects are included is shown in plots 3 and 4. The availability of the two-of-three system, with a spare and using λ_T, actually decreases slightly over the 20 year span but this cannot be seen in the scale of Fig. 5.

Observations

This work has focused on updating methods for failure rate estimation including the presence of aging effects. This leads to more complexity in the analysis of raw data and in the reliability and safety analyses which follow.

Some of the possible effects on reliability and availability have been demonstrated in the Results section. For some systems, the aging effects will be less and for others, greater. It would seem to be an appropriate time to continue major efforts in the study of aging effects and the further development of methods for failure rate estimation.

Conclusions

The presence of aging effects and the resulting time-dependent failure rates have greatly increased the need for enhancing the scope and the level of detail of data collection in the field. Now, more attention must be given to environmental and stress factors when collecting and evaluating field data. Some of the methods mentioned above should be utilized to a greater extent than in the past. These and other methods should be developed further to obtain better estimates of the aging and random portions of the total failure rate. An increase in data collection, in-service inspection and maintenance procedures will provide valuable information which should be factored into the design of mechanical components rather than after the fact.

The design process of mechanical components should place more emphasis on materials and the environment pertaining to the intended application. Data bases for mechanical components should be expanded to include stress factors similar to those for electronic components (HDBK-217D). This, of course, will be more difficult but is no less necessary than for electronic parts. Data collection systems in the utilities should be capable of evaluating data, estimating failure rates, uncertainties, and combining generic and sparse plant-specific data where necessary. These data base systems should be able to identify and quantify aging rate phenomena, handle censored data and estimate confidence limits. Newer methods such as those above, should be explored further to improve the quality of the failure rate information since this serves as the input to many critical economic and safety analyses.

Acknowledgements

The work presented in this paper constitutes part of a program performed under contract to Northeast Utilities Service Co. The authors appreciate the encouragement and support of P.M. Austin, NU. The contributions of F.O. Cietek (now at Texas Utilities) in the early phase of this work are gratefully acknowledged.

References

1. Vesely, W.E., "Risk Evaluations of Aging Phenomena: The Linear Aging Reliability Model and Its Extensions," NUREG/CR-4769, April 1987.

2. Apostolakis, G., "Bayesian Methods in Risk Assessment," Adv. in Nucl. Sci. & Tech, 13, 415 (1981).

3. Kaplan, S. "On the Method of Discrete Probability Distributions in Risk and Reliability Calculations -Application to Seismic Risk," Risk Analysis, 1, No.3, 189 (1981).

4. Martz, H.F., "An Error and Uncertainty Analysis of Classical and Bayesian Failure Rate Estimates from a Poisson Distribution," NUREG/CR-2465, January 1982.

5. Vesely, W.E., et al, "A Bayesian Analysis of Diesel Generator Failure Data," NUREG-0873, January 1982.

6. Heising, C.D., and Patterson, E.I., "Plant Specification of Generic Human-Error Data Through a Two-Stage Bayesian Approach," Rel. Eng. 7, 21 (1984).

7. Nelson, W., "Hazard Plotting for Incomplete Failure Data," J. Quality Tech. 1, 1, 27 (1969).

8. Nelson, W., "Hazard Plotting Methods for Analysis of Life Data with Different Failure Modes," J. Quality Tech. 2, 3, 126 (1970).

9. Nelson, W., "Weibull Analysis of Reliability Data with Few or No Failures," J. Quality Tech. 17, 3, 140 (1985).

10. Lawless, J.F., "Statistical Methods in Reliability," Technometrics 25, 4, 305 (1983), a review paper with valuable critiques following.

11. Thompson, Jr., W.A., Discussion article of Ref. 10, Technometrics 25, 4, 326 (1983).

12. Vaurio, J.K., and Linden, G., "On Robust Methods for Failure Rate Estimation," Rel. Eng. 14, 123 (1986).

13. Hahn, G.J., and Nelson, W., "A Survey of Prediction Intervals and Their Applications," J. Quality Tech 5, 4, 178 (1973).

14. Nelson, W., Applied Life Data Analysis, Wiley, 1982.

A TIME DEPENDENT METHODOLOGY FOR EVALUATING COMPONENT RELIABILITY

D.L. Sanzo and G.E. Apostolakis
School of Engineering and Applied Science
University of California

Abstract

A methodology has been developed to assess component lifetime. This methodology integrates both normal and transient (abnormal) operating conditions through thermophysical models, as well as the attendant uncertainties. This results in a set of reliability curves for the component.

The methodology developed consists of the following steps: 1) Identifying those particulars deemed important in characterizing normal and transient operation. 2) Failure Modes (FM's) and associated Failure Criteria (FC) to be included in the analysis. Previous lifetime studies, existing design codes, and engineering judgment are used as guidelines in choosing relevant FM's and FC. 3) Model development for normal operation of FM's. This is one of the most difficult parts in assessing component reliability due to the complexity of the phenomena involved. 4) Assessment and propagation of uncertainties and generation of reliability curves. After the development of the appropriate failure mode models, expressions for the time to failure under each FM are generated. Due to common dependencies, all environmental and loading effects on times to failure for each FM must be considered jointly. Once the times to failure are obtained, they can be combined to obtain families of reliablity curves.

1. Introduction

One of the key areas surrounding the successful continuation of nuclear power is reliability. As nuclear plants age, concerns about reliability intensify. The impact of aging is seen via component reliability. Indeed, the reliability of particular components and systems has been recognized as a key attribute in determining the economic feasibility and safety risk of power plants.

Given that reliability is a key factor in determining the viability of nuclear power, substantial effort has been expended to acquire the information needed to assess the reliability of reactor components and systems. This effort has proceeded along several fronts which can be summarized as follows:

1. Design of entire nuclear reactors.
2. Deterministic lifetime prediction of specific components which includes only normal operation and associated failure modes, i.e., transient (abnormal) events are excluded, with some attempts at showing the effects of uncertainties.
3. Deterministic lifetime prediction of specific components based in part upon a fixed quantity associated with transients and associated failure modes, with some attempts at showing the effects of uncertainties.
4. Specification of allowable parameter values to meet particular design or failure criteria.
5. Identification and analysis of transient events and their frequency of occurrence.
6. Data collection, analysis, and failure rate prediction and reliability assessment for reactor components and systems, under non-transient operating conditions.
7. Model development for phenomena peculiar to normal reactor operation and reactor transients.
8. Testing, selection, and behavior prediction of reactor materials.

It is worth noting that a comprehensive reliability assessment would allow all the above items to be combined in a consistent fashion. Specifically, the ability to calculate component reliability with aging effects requires that all of these pieces be combined in a coherent manner, allowing for possible synergism between normal operations and transient events.

With this in mind, it is the objective of this research to implement a methodology that will provide the needed framework that allows both normal operating conditions and transient events along with their attendant uncertainties to be combined, such that time dependent component reliability can be assessed.

The remainder of this paper is organized as follows: in Section 2, the methodology to assess component reliability is presented; in Section 3, some results from implementing this methodology for a tokamak fusion reactor component known as the limiter are presented; and Section 4 offers some comments and conclusions.

2. Methodology

2.1 Reliability

Since the goal is to assess component reliability, a method must be developed for use in achieving this goal. As a first step, several terms of interest must be precisely defined. Reliability, as used in this work, means: the relative fraction of time the component performs its intended function without failure. It is a probabilistic concept. By failure, it is meant that a component has become unacceptable under a given set of Failure Criteria (FC), via a postulated set of Failure Modes (FM's). The FM's and FC which must be considered are based both on judgment, as in the case of theoretical studies and experience, as dictated by design codes which have evolved over time.

The important elements in assessing time-dependent component reliability are the environmental conditions under which the component is operated and its load history. If the component is expected to operate at elevated temperatures both during normal operation and transient conditions, or in a highly irradiated environment, it is to be expected that materials properties and the component's history are the key factors in assessing its reliability.

2.2 Discussion of Steps Required to Assess Reliability

A block diagram of the methodology used to obtain component reliability is shown in Fig. 1. This methodology is broadly based on techniques developed to perform Probabilistic Risk Assessments (PRA) of fission nuclear power plants.[1]

Briefly stated, the PRA approach is centered about the answers to the following questions[2]:

1. What can happen (i.e., what are the scenarios)?
2. How likely is it that it will happen?
3. If it does happen, what are the consequences?

PRA addresses the issues and ramifications of uncertainty via the probability-of-frequency approach.[2-3] This involves both classical and Bayesian interpretation of probability. The classical, or frequentist, interpretation is used to characterize statistical or stochastic uncertainty, while the Bayesian, or subjectivistic, interpretation is used to describe state-of-knowledge uncertainties. This approach allows a more complete picture of component reliability to be drawn. In essence, there is obtained a set of curves that expresses the uncertainty in the reliability value at a given point in time. The blocks in Fig. 1 are examined in more detail below.

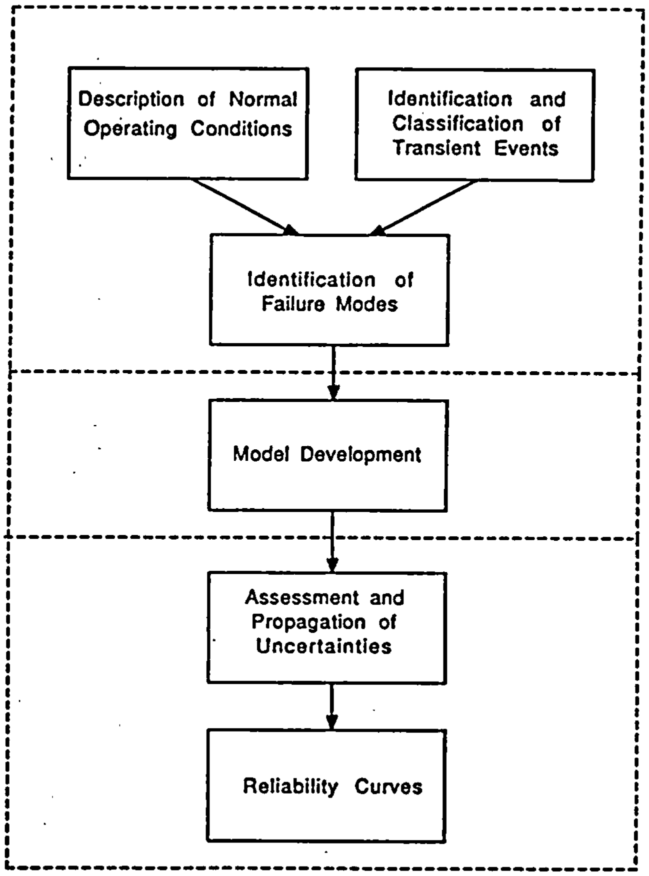


Figure 1. Methodology for Analyzing Component Reliability.

2.3 Normal Operating Conditions

Normal operating conditions depend on many factors. These include such items as: the type of reactor (BWR, PWR, etc.), the materials used in designing the reactor (SS, Graphite, etc.) and the layout and design of required subsystems. The particular reactor characteristics determine such items as normal temperature gradients and neutron fluxes through components.

2.4 Identification and Classification of Transient Events

The identification of Initiating Events (IE's) which might affect component reliability can be obtained via a Master Logic Diagram (MLD) or any other standard method.[1] IE's are events capable of initiating transients, e.g., coolant pump failures and valve failures.

Once the IE's have been identified, a classification scheme for the IE's based upon their effects on the component is required. The desired results of such a classification are the types of transient events, their frequency of occurrence and their impact on the component.

Finally, normal operating conditions and transient events must be considered in conjunction for several reasons, including (assuming these FM's are applicable):

1. Fatigue and creep damage are cumulative effects. Since they depend (in part) on the difference between the maximum and minimum stress in a cycle, the occurrence of a transient event during a particular cycle will impact this difference. Furthermore, the number of transient events which occur is important.
2. Depending on the severity of a transient event, large thermal strains and thermal creep may occur, all of which impart a residual stress distribution that affects the stress history of the component.
3. The particular time when a transient event occurs is of importance because material effects occur over different time scales, e.g. irradiation-induced material changes such as embrittlement. Thus, a transient occurring at the beginning of operation may have a different impact than one occurring later.

2.5 Identification of Failure Modes and Specification of Failure Criteria

In deciding what Failure Modes and associated Failure Criteria are to be included in the analysis, previous studies, design codes and engineering judgment are used as guidelines. The identification of failure modes is tied in with the MLD via the undesirable top event. Since the top event eventually determines the IE's, it is clear that the identification of FM's is related to the IE's. The process of identifying FM's and IE's is an iterative procedure. Thus, in Fig. 1, the order of the methodology should not be taken as strictly top down.

After identifying the FM's, what constitutes failure must be decided. This involves specifying some adequate and proper failure theory that takes into account how the various damage mechanisms causing degradation interact. It also includes specifying limits, i.e., FC, on the appropriate parameters describing the failure theory.

2.6 Model Development

This is one of the most difficult parts in assessing component reliability due to the complexity of the phenomena involved. Judgment is required in deciding what level of sophistication is required in developing models. It is of little use to develop a particular model that is extremely accurate, while another model that is relatively inaccurate influences the results to a far greater degree. The temptation to overanalyze phenomena that are understood, relative to analyzing phenomena that are not understood must be avoided. This is not a trivial task. Models germane to reactors would include, for example:

1. Thermal models to calculate temperature distributions through a component during normal and transient conditions;
2. Stress-Strain models incorporating thermal and irradiation creep, as well as swelling;
3. Failure Mode models for failure due to cumulative cyclic damage and strain limits; and others as needed.

An example of a failure model is as follows (for details see Ref. 4). Under the cumulative cyclic damage FC, the number of cycles until the damage threshold, D, is reached depends upon the number of transients that have occurred. Each transient causes an incremental amount of damage, $\frac{1}{N_{TF}}$. The component fails when the accumulated damage exceeds D. The survival frequency can be written as

$$fr(T_F \geq t) = fr\left(\frac{k}{N_{TF}(t)} \leq D\right) \tag{1}$$

where T_F is the component failure time, k is the number of transients up to t (a random variable), and

$$\frac{1}{N_{TF}(t)} = \frac{\frac{\sigma_p}{E}\left(\frac{\sigma_t'}{\sigma_y(t)} - 1\right)}{\epsilon_{up}(t)} + \left(\frac{\sigma_t' - \sigma_y(t) + \sigma_p}{1.25\ \epsilon_{up}(t)}\right)^{1.25} \tag{2}$$

We note that the incremental amount of damage, $\frac{1}{N_{TF}(t)}$, depends on time through the yield stress, $\sigma_y(t)$, and plastic ductility, $\epsilon_{up}(t)$. These are functions of time because of irradiation, which leads to an overall increase of $\frac{1}{N_{TF}(t)}$ with time.

2.7 Assessment and Propagation of Uncertainties and Generation of Reliability Curves

Several steps are required to obtain reliability curves and to extract useful information. Some of the important aspects are discussed below.

After the development of the appropriate failure mode models, expressions for the time to failure under each FM are generated. Due to common dependencies, all environmental and loading effects on times to failure for each FM must be considered jointly. Once the times to failure are obtained, they can be combined to obtain component reliability. The probability-of-frequency approach then allows a set of relibilility curves to be generated.

Note that reliability is a conditional function of state-of-knowledge variables[3]; thus, to quantify the reliability, a particular set of values for the variables must be chosen. This implies that the reliability at a given point in time actually has a probability distribution associated with it. This probability distribution is directly attributable to state-of-knowledge uncertainties in the variables and models of which the reliability is a function. Thus, a set of curves is developed by allowing all variables and models which have state-of-knowledge uncertainties to take on their range of values and then calculating the reliability based on the particular set of values chosen, i.e., the uncertainties are propagated. The resultant reliability values then form a probability distribution at each point in time.

The range of values which a particular variable or model might have can be obtained from: expert opinion, existing reactor analyses, data developed from reactor experience, and other engineering analyses as required. For example, the damage threshold of Eq. (1), D, is not well known. Using available recommendations by experts and actual observed data, we express our uncertainty about this value by a lognormal distribution with the following characteristic values: mean: 1.00; median: 0.78; 5th percentile: 0.25; 95th percentile: 2.47 (for details, see Ref. 4).

3. Limiter Reliability

The methodology discussed here has been applied by the authors to analyze a tokamak fusion reactor component known as the limiter[4]. The limiter was chosen for analysis because of its considerable importance in tokamak fusion reactors. Its basic function is plasma impurity control. In performing its function, the limiter is exposed to extremely high heat and particle fluxes in addition to being thermally cycled between "on" and "off" conditions. Phenomena relevant to normal operation include erosion, swelling, irradiation creep, thermal fatigue and irradiation induced materials changes.

Transients known as "disruptions" can occur. These transients can deposit a large amount of energy on the limiter over a small period of time, causing a large heat flux to be imposed on the limiter. Attendant phenomena for transients include melting and vaporization, thermal creep and plasticity effects. More details can be found in References 4 and 5.

Shown in Fig. 2(a) are the 10th, 50th and 90th percentiles of reliability for the baseline INTOR limiter under the assumptions in Ref. 5. These assumptions include: a 1% irradiation creep strain limit; a 5% swelling strain limit; and material loss during disruptions due to both melting and vaporization. In addition, transients occur relatively frequently. The dominant FM in this case is cumulative cyclic damage, i.e., effects due to creep-fatigue interactions during normal operation and transients. Note that curves for various percentiles essentially fall on top of one another for some time values. This is due to very short lifetimes predicted under irradiation in conjunction with limiting values of thermomechanical properties under irradiation. However, at some point in time it can be seen that the probability distribution for limiter reliability has a spread. This spread is caused by uncertainties in the parameters characterizing cumulative cyclic damage and the frequency of transients.

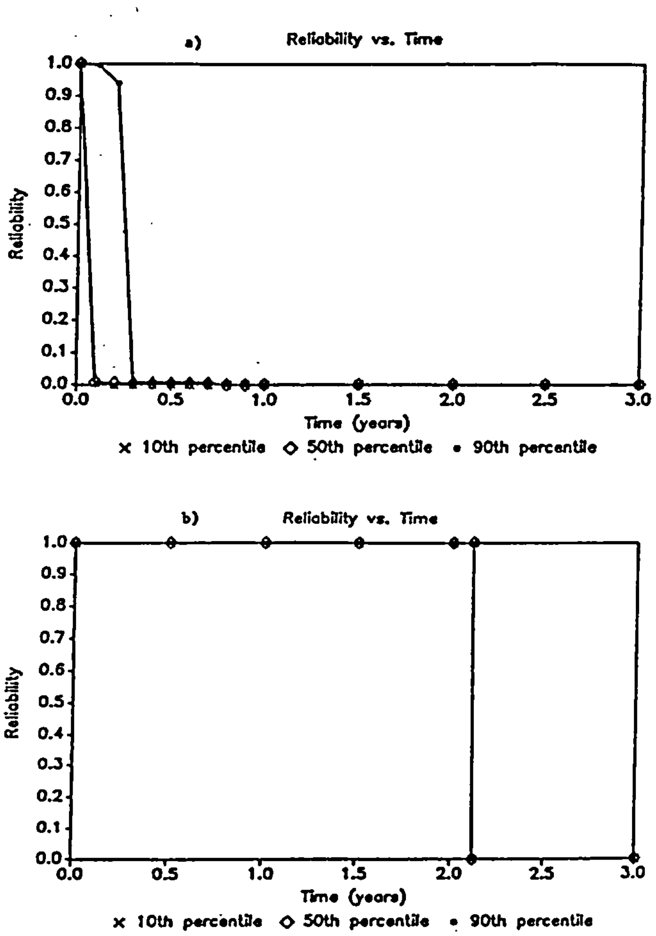


Figure 2. INTOR Limiter Reliability vs. Time[5]

If the INTOR reactor could be designed such that no transients of any type occurred, a situation shown in Fig. 2(b) would result. This group of curves shows that limiter reliability would be perfect until erosion of material during normal operation resulted in failure. When compared with Fig. 2(a), the impact of transients on limiter reliability is obvious. The curves in this case exhibit small uncertainty, since most of the uncertainties were related to parameters characterizing transients.

4. Conclusion

A methodology has been implemented which allows both normal operation and transient events to be combined for component reliability prediction. This methodology is time-dependent, thereby allowing phenomena that impact component aging to be incorporated into the analysis. The impact of aging is seen via component reliability. The methodology was used to assess the reliability of a tokamak fusion component, the limiter.

References

1. American Nuclear Society, *PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessment for Nuclear Power Plants, 1&2,* NUREG/CR-2300 (1983).
2. S. Kaplan and B.J. Garrick, "On the Quantitative Definition of Risk," *Risk Analysis*, 1:11-28 (1981).
3. G. Apostolakis, "Uncertainty in Probabilistic Safety Assessment," *Trans. 9th Int'l. Conf. on Structural Mechanics in Reactor Tech.*, Vol. M:395-401, Lausanne, Switzerland, Aug. 17-21, 1987, A.A. Balkema Publishers, Boston.
4. D.L. Sanzo and G.E. Apostolakis, "Limiter Probabilistic Lifetime Analysis," accepted for publication in *Fusion Engineering and Design*.
5. W.M. Stacey, Jr., et al., *INTOR*, USA INTOR/85-1 (1985).

APPLICATION OF STRUCTURAL RELIABILITY AND RISK ASSESSMENT TO THE MANAGEMENT OF AGING

T. A. Meyer

K. R. Balkey and B. A. Bishop

Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

and

M. F. Moylan

Wisconsin Electric Power Company
Milwaukee, Wisconsin

INTRODUCTION

There can be numerous uncertainties involved in performing component aging assessment in an effort to manage component aging. In addition, sufficient data may be unavailable to make a useful aging prediction. Structural Reliability and Risk Assessment (SRRA) is primarily an analytical methodology or tool that quantifies the impact of uncertainties on the structural aging of plant components and can address the lack of data in component life prediction.

As a prelude to discussing the technical aspects of SRRA, a brief review of general component aging and life prediction methods is first made so as to better develop an understanding of the role of SRRA in such evaluations. SRRA is then presented as it is applied in component aging evaluations with several example applications being discussed for highest priority components as defined by PWR Life Extension (PLEX)* programs.

GENERAL COMPONENT LIFE PREDICTION METHODS

In a power plant, there is a broad range of issues and concerns regarding the prediction of component aging that must be addressed. Some of the pertinent issues and concerns are:

- o Determination of degradation mechanisms
- o Loss of function versus loss of structural integrity
- o Predictive methods to be utilized in the aging assessment
- o Procurement of inspection and maintenance program data
- o Definition of equipment and components of concern
- o Uncertainties

The last item above is identified simply as "uncertainties" and refers to those uncertainties that potentially exist in the preceeding five entries of the list. Such uncertainties include: the degree of importance of particular degradation mechanisms, range of load or material properties affecting structural integrity, the variance inherent in the predictive methods, the uncertainties in flaw characterization and timing of inspection programs, and ultimately the cumulative effect of all of these uncertainties on prioritizing and managing the aging of components of concern.

An essential ingredient of an aging management program is the procurement of pertinent data required to assess the impact of aging on component life. Data typically required in such an assessment are:

- o Normal and transient loads
- o Environmental conditions
- o Prior inspection results
- o Design and fabrication records
- o Failure observations and material properties
- o Component and system functional requirements

---

* The term "Plant Life Extension (PLEX)" as used in this paper originates from the definition of "Life Extension" contained in the Atomic Industrial Forum/National Environmental Studies Project-040 (AIF/NESP-040) and shall not be deemed to be a warranty or representation that such plant and/or equipment can be operated economically and safely for the initial license term or beyond. Rather the term "PLEX," which is based upon engineering judgements, technological developments, operating plant data and analysis of such developments and data attempts to predict the capability for continued operations of the plant and for the equipment beyond the licensing basis of such plant and/or equipment.

These data are essential to the resolution of aging management and life prediction issues and concerns identified previously, especially the definition of uncertainties, regardless of the technical tools applied in the assessment.

A number of life prediction methods and tools have been and are being utilized to assess component aging. These tools can be separated into the two broad categories of "Historical" and "Current and Developing" methods.

Figure 1 gives a graphical summary of "Historical" methods, typically employed for component life prediction. Figure 1 divides the historical methods into: 1) those that are directed at evaluation of aged materials behavior, 2) those directed at analytically predicting component loss of function or structural integrity, and 3) those aimed at an operator's perspective of what is or is not done to equipment in the field to affect component aging. In each of the subcategories in Figure 1, time lines show the emergence of various life assessment methods and the relative timing of their formulation.

A graphical summary of "Current and Developing" methods is given in Figure 2 which divides the methods into "Deterministic" and "Probabilistic" methods and relates the various methods to each other on the basis of their relative maturity.

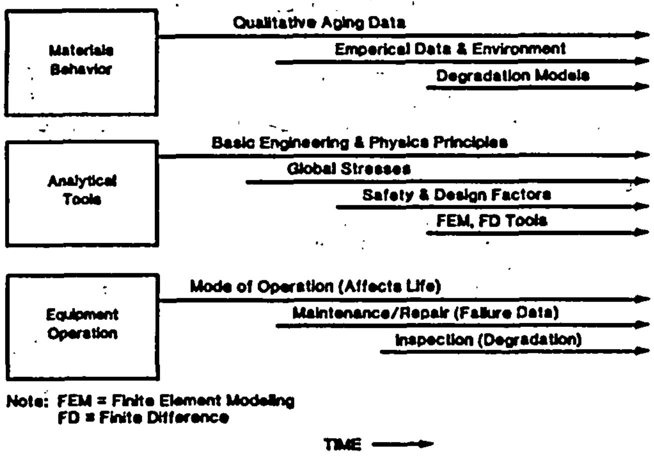


Figure 1 -- Historical Life Prediction Models.

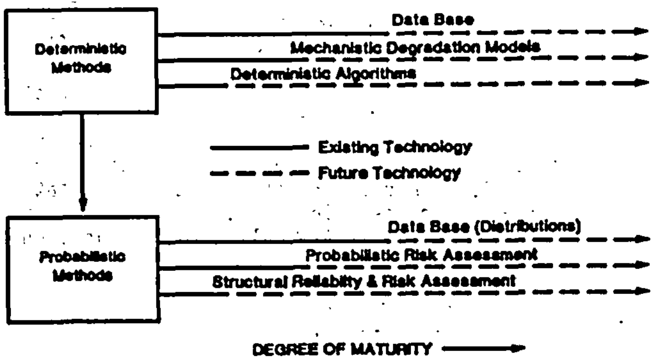


Figure 2 -- Current and Developing Life Prediction Models

Deterministic life assessment methods employ conservative data and assumptions in mechanistic degradation models or life prediction algorithms with the outcome typically being a very conservative result. If a conservative result is acceptable for the situation being evaluated, then the uncertainties bounded in the assessment pose no concern for the management of aging. However, as equipment ages, there is a growing, and often times unacceptable, consequence of bounding uncertainties in the assessment of component life. In many cases, insufficient data exist to make a prediction using standard deterministic methods. In those cases where conservative results are not acceptable or data is insufficient, probabilistic methods are an attractive alternative or supplement to the deterministic life assessment process. Probabilistic methods have the ability to explicitly consider uncertainties, including the quantitive effect of them on component aging, or can make predictions where a dearth of data exists.

With this background, the technical aspects of probabilistic methods in general and SRRA in particular will be described, letting the prior discussion of component aging and life assessment methods serve as a demonstration of the role that probabilistic tools can fulfill in life extension efforts.

SRRA AS APPLIED IN COMPONENT AGING AND RESIDUAL LIFE EVALUATION

The primary objectives of probabilistic engineering methods are to address and quantify uncertainties and to provide additional bases for effective decision making. An additional perspective of the benefits of probabilistic methods is that "Probabilistic methods can move the assessment of risk (both economic and technical) from the hands of the analyst, who chooses the bounds for the uncertainties in a deterministic assessment, to the hands of the appropriate decision makers within a utility." The significance of this process is that typically a utility decision maker has no way of knowing the degree of risk inherent in a deterministic aging or life prediction whereas the risk is explicitly provided in a probabilistic assessment.

Expanding upon the prior definition of SRRA, it is an analytical methodology or tool that combines traditional deterministic structural analysis techniques with probabilistic methods to assess aging and determine component reliability. This determination of reliability is accomplished by quantifying the impact of uncertainties in materials, loads and flaws on the structural life of components and by addressing the lack of data in component life prediction. A more extensive treatment of probabilistic methods relative to structural integrity is given in Reference 1.

As shown in Figure 2, Probabilistic Risk Assessment (PRA) is an additional probabilistic tool that is being employed in life prediction evaluations. PRA utilizes historical failure and reliability data in conjunction with event tree and fault tree methods to predict the life of a system and/or components.

The primary distinctions between PRA and SRRA are:

Probabilistic Risk Assessment

- o Utilizes historical failure and reliability data
- o Utilizes event tree and fault tree methodologies
- o Focus is primarily on a system or overall plant level (macroscopic)
- o Does not explicitly consider specific aging mechanisms

Structural Reliability and Risk Assessment

- o Addresses the structural reliability of a component
- o Quantifies uncertainties in materials, applied loadings and flaws
- o Typically utilizes Monte Carlo simulation techniques with mechanistic models
- o Can provide failure probability values as inputs to a PRA when historical failure data is limited
- o Focus is primarily on a component or structural failure mode (microscopic)
- o Can explicity consider the effect of specific aging mechanisms

PRA and SRRA have been used as independent reliability tools, and they have also been used together in life assessment evaluations. These tools are cost-effective today for addressing complex issues because of rapid improvements over the past decade in computer and analytical efficiency, such as the use of importance sampling in SRRA applications (1).

Example applications of SRRA to aging and life assessment are given in the following sections of this paper and include a case where SRRA results have been coupled together with PRA results.

HIGHEST PRIORITY COMPONENT AS DEFINED BY PWR PLEX (PLANT LIFE EXTENSION) PROGRAMS

Several studies have been performed to date to evaluate the potential and required actions to extend the life of PWR plant systems. Full scope or partial (component or individual system) life extension programs have been done for at least six plants and the basic approach typically includes the following considerations.

- o development of new record keeping methods designed to track key parameters for prediction about aging and life span
- o plant-specific evaluations of major systems and components to determine (a) how they are aging, (b) the availability of spare parts for them, and (c) whether future refurbishments will be needed, including what type are required
- o improved or augmented maintenance programs for critical components
- o development of a component prioritization list to define the key components critical to future life extension. The prioritization process typically rates plant systems and components against factors including cost of replacement, impact on downtime, impact on safety and regulatory importance.

Even though these PLEX studies have had a goal of demonstrating the potential to extend the plant life beyond its original design basis, the methods and results are readily applicable to "Management of Aging" in nuclear power plants. This relationship is a result of the underlying aging evaluation performed for any Plant Life Extension program considering the definition of pertinent aging mechanisms, modeling those mechanisms and evaluating the impact of those aging mechanisms on system or component life. Typical results of a Plant Life Extension program include the definition of actions that can be taken to either minimize or control the aging process and/or to monitor and better know the aging critical plant systems and components.

Both the PLEX aging evaluations and the associated results in terms of controlling, understanding and minimizing component and system aging are pertinent and valuable to the "Management of Aging" in nuclear power plants. For this reason, this paper will utilize a list of highest priority components and pertinent aging evaluation studies taken from prior PLEX studies to identify several components/systems to serve as example applications of SRRA to the "Management of Aging."

A completed portion of the ongoing PLEX evaluation of SURRY 1 provided the "PLEX PRIORITIES LIST" (2) as shown in Table 1.

TABLE 1
PLEX PRIORITIES LIST (2)

RANKING	COMPONENT/STRUCTURE
1	*Reactor Vessel
2	*Containment
3	Reactor Vessel Supports
4	*Reactor Coolant Piping
5	Steam Generator
6	Emergency Diesel Generators
7	*Reactor Vessel Internals
8	Reactor Coolant Pump Casing
9	Pressurizer
10	Neutron Shield Tank

These components/systems were ranked and prioritized utilizing the factors previously defined in this paper for this purpose. The components/systems having an asterisk (*) are to be utilized in the succeeding sections of this paper as example applications of SRRA to the Management of Aging. Some of the information provided will be taken from actual aging studies performed on the component/system and other information will be provided that is intended to give insight as to how SRRA may be applied to the Management of Component Aging for cases not previously evaluated.

REACTOR VESSEL

In the development of a matrix of potential degradation mechanisms for reactor vessel components in Reference 2, thirteen subcomponents of the reactor vessel and nine potential degradation mechanisms were identified and ranked in terms of level of concern. The levels of concern were ranked in terms of low (>80 years life, no concern to PLEX), medium (60-80 years of life, some concern to PLEX) and high (<60 years of life, should be evaluated for PLEX). Within the entire potential degradation matrix defined above (117 combinations), five combinations of subcomponents and potential degradation mechanisms were defined as medium to high level of concern and only one matrix location was defined as a high level of concern.

The one high level of concern was the "Intermediate and Lower Shells" subcomponent in combination with radiation induced embrittlement.

Given this high level of concern for the embrittlement of the intermediate and lower shells of the reactor vessel, the following paragraphs provide an example of prior work that has been performed utilizing SRRA in assessing the impact of embrittlement on reactor vessel integrity.

The most extensive application of SRRA coupled with PRA to date has been in the determination of the risk of significant flaw extension in a major nuclear reactor pressure vessel.

In the late 1970's, several plant operating events occurred that resulted in a rapid and severe cooldown in the primary reactor coolant system coincident with a high or increasing primary system pressure. These events were termed pressurized thermal shock (PTS) transients. A concern arose that such events may induce the propagation of a flaw assumed to exist in the reactor vessel wall, thereby potentially affecting the integrity of the vessel. The concern was directed toward the beltline region of pressurized water reactor (PWR) vessels since reduced fracture resistance may exist because of irradiation induced embrittlement and this portion of the vessel would be subjected to PTS.

To respond to this concern, an innovative technology was developed that coupled results from PRA event sequence analysis, which took into account potential plant operator errors and system malfunctions, with results from thermal-hydraulic and SRRA (probabilistic fracture mechanics analyses) to identify all potential transient scenarios of concern and to determine the risk of significant flaw extension in the reactor vessel from PTS events as shown in Figure 3. The SRRA models included distributions for important parameters where little or no data existed, such as the potential presence of near surface flaw indications. (Data from non-nuclear vessels were used.)

This probabilistic methodology, which is required as part of future plant-specific evaluations of vessels that are projected to exceed NRC requirements on the issue (3), is also a very powerful tool for gaining insight into the management of reactor vessel aging relative to embrittlement.

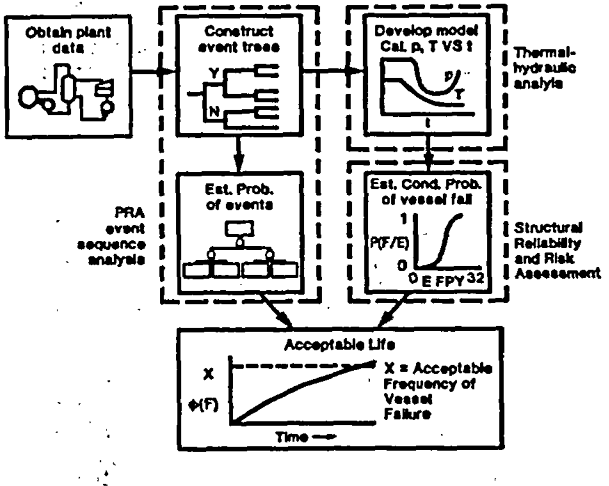


Figure 3 -- Flow Chart to Evaluate the Risk of Significant Flaw Extension in a Reactor Vessel for Pressurized Thermal Shock

This can be accomplished by referring to additional prior applications in the context of aging management. In one case (4) SRRA results coupled with PRA results were utilized to provide a basis for quantifying the aging or embrittlement of the reactor vessel as well as quantifying the effect of various utility initiatives on reducing the "frequency of significant flaw extension" in the reactor vessel.

Figure 4 provides the frequency of dominating transient occurrence as a function of age for significant flaw extension using deterministic fracture mechanic results in a stepwise continuous plot (4). The plot is a summation of the individual results (upper plot). The definition of dominating transients was obtained from probabilistic fracture mechanics (SRRA) and PRA results that identified the set of transients that dominated the overall risk of significant flaw extension in the reactor vessel wall. The "PTS Safety Goal" can be considered to be an "acceptable frequency of occurrence" of PTS events and forms a lower boundary of the "Region of Regulatory Concern." As long as the stepwise continuous plot does not enter this defined region, it is considered that the accumulated risk is less than a value that would be of concern to the regulatory bodies. Figure 4 also provides a lower stepwise continuous plot that shows the impact of plant-specific initiatives for reducing the frequency of significant flaw extension. It is this curve that provides the ability to "manage" the aging of the rector vessel in terms of allaying reactor vessel integrity concerns that develop relative to PTS in a cost-effective manner.

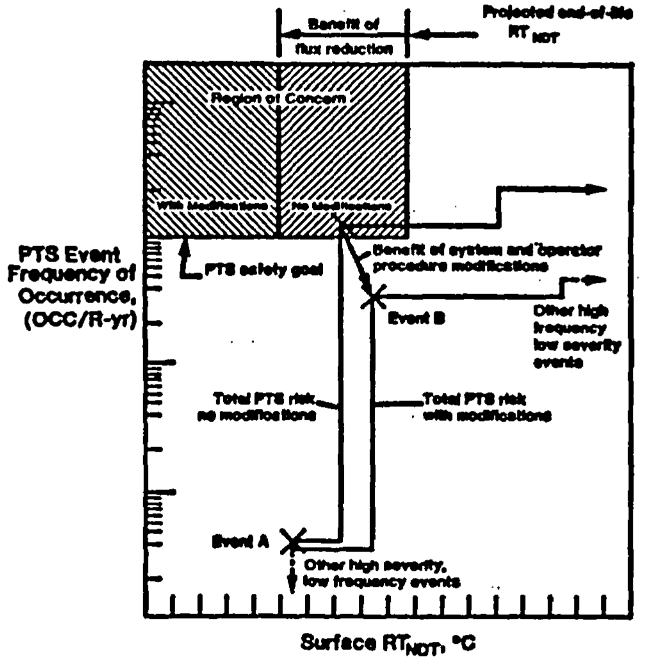


Figure 4 -- Impact of Plant-Specific Measures on the Risk of Significant Flaw Extension in a Vessel for Decision-Making for the Management of Aging

CONTAINMENT

Probabilistic methods can be applied to evaluate containment integrity. The application of probabilistic methods (PRA & SRRA) to a containment is more complex than the application of these tools to reactor vessel PTS because of the diversity of the containment structure. Unlike the work done to address PTS, which utilized a well-characterized damage model (neutron embrittlement), a well-founded failure methodology (linear elastic fracture mechanics), and a homogenous structure and vessel location (beltline wall), a containment structural risk analysis must include several structural components, multiple critical locations and geometries, a number of failure models, and a number of damage mechanisms, some that are location specific. The matrix in Table 2 contains some of what might be considered for an SRRA evaluation of a post-tensioned concrete containment building:

TABLE 2
POSSIBLE APPLICATIONS OF SRRA
TO CONTAINMENT COMPONENTS

COMPONENT	POTENTIAL CRITICAL LOCATIONS	POSSIBLE DAMAGE MECHANISMS
Concrete	-Ring guider -Tendon gallery -Basemat -Exterior surfaces	-Freeze-thaw damage -Aggregate reactions -High temperatures -Calcium hydroxide leaching -Corrosion of embedded steel -Aggressive chemical attack -Corrosion due to soil or groundwater characteristics -Irradiation -Fatigue -Construction practices
Liner Plate	-Below grade liner plate -Floor liner -Near penetrations	-Corrosion -Physical abuse -Spills -Elevated temperature -Irradiation -Fatigue

Additional structural components that should be included in an overall SRRA are piles, the pre-stressing system, hatches, piping and electrical penetrations, and other structural and reinforcing steel (see Figure 5).

As can be seen for a containment building, the number of structural components and damage mechanisms that need to be considered increases over those considered for reactor vessel integrity. The method to address these mechanisms, however, does not change. The power of probabilistic methods is that all of these variables can be statistically combined into an overall containment failure probability model.

To determine the probability of a major containment failure, SRRA results need to be coupled with PRA results. Similar to the work done for PTS, event trees can be constructed that yield a comprehensive set of transient loadings and the estimated frequencies of these events. This process would also result in a diverse set of events and would include those from:

- o 10CFR50 Appendix J - Integrated leak rate testing
- o Design basis transients - Loss-of-coolant-accidents and steamline breaks
- o Appendix A loads - Fire, flood, seismic, tornado, etc.
- o Plant operations, modifications, and maintenance activities.

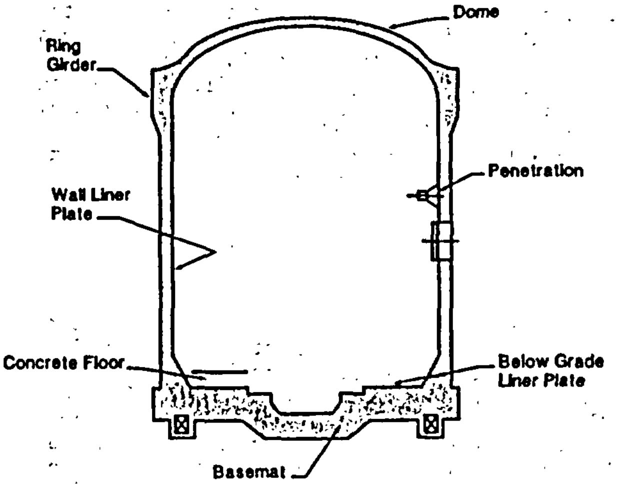


Figure 5 -- Containment Components With Critical Locations Identified

As in the SRRA methodology used for PTS, the containment transient loads are applied to each component and location. The parameters pertinent to the damage mechanisms are fitted into distributions and treated as random variables. For purposes of assessing failure, most of the aging mechanisms can be assumed to cause the formation of cracks, a reduction in cross-sectional area, a reduction in physical strength, and/or physical discontinuities. These parameters can then be utilized in the appropriate probabilistic failure (fracture mechanics, general yield) models to determine the component risk of failure.

Of course, failure for some structural components does not always mean that containment has lost integrity. There is some redundancy. For simplicity, however, this could be assumed, and the resulting failure probabilities from each structural component and location summed to obtain the overall risk of containment failure. The process would identify which components are most important for further evaluation.

Both the overall containment failure model and the individual component analyses that make up that model have utility. The summation of results may be input for other probabilistic safety and consequence studies. The results also may be used for current licensing and relicensing (PLEX). They provide an indication of how containment aging mechanisms affect the containment failure probability with time, something that containment testing cannot indicate.

The results from the individual failure analyses that make up the overall model provide additional, detailed decision-making information. This information includes decisions for inspection frequency, maintenance, or surveillance to obtain more data on the manifestation of the aging mechanisms, particularly corrosion, on certain components. Application of probabilistic methods to assess containment integrity has merit even if it is only applied to a single structural component in containment, because it provides a means to deal with uncertainties and lack of hard data, especially with respect to the aging mechanisms and therefore the management of aging.

REACTOR COOLANT PIPING

A significant amount of probabilistic work has been done to address the integrity of reactor coolant piping as well as other major piping systems for both PWR and BWR applications.

For PWR systems, both seismic event PRA and SRRA methods have been applied in the evaluation of leak-before-break for the primary piping (5,6,7). The basic concern was that an earthquake could cause a large primary pipe rupture, resulting in an inability to adequately cool the reactor core. The leak-before-break concept is a demonstration by analyses that a piping system can tolerate large flaws, that such flaws, upon becoming through-wall, are stable and will not extend significantly, and that the resulting leakage will be detected so that appropriate actions can be taken. (Some recent pipe cracking incidents in large - 6" or greater - piping systems support the validity of this concept.) The earliest probabilistic evaluation addressed uncertainties in loading, fracture toughness, crack existence and crack extension (5) to demonstrate that the probability of failure was extremely low and that a leak would occur before a break. The latter independent evaluations (6,7) confirmed the earlier results leading to the regulatory acceptance of the leak-before-break concept (8).

Relative to the management of aging for primary piping systems, probabilistic work is currently underway to address concerns associated with degradation mechanisms such as thermal aging. This phenomenon has only been recognized within the last decade as occurring in cast stainless steels at operating temperatures of nuclear reactors. Useful material test data has become available only very recently for making better predictions. The material behavior is synonymous to the irradiation embrittlement of reactor vessel material in that the fracture toughness behavior is dependent upon temperature and time. Only probabilistic methods such as SRRA can address uncertainties associated with this aging mechanism.

However, as a result of the recent cracking incidents mentioned previously, a strong impetus exists for using more extensive PRA and SRRA methods to better manage aging in all major piping systems for the long term. Cracking occurred as a result of unanticipated thermal stratification and thermal striping in the affected piping systems. More research is needed using risk methods, such as properly executed failure modes and effects analyses, that can better pinpoint potential system and component situations that may lead to unanticipated loading conditions including those with dynamic effects. For critical locations, SRRA evaluations could be performed to define cost-effective inspection and monitoring programs for the management of aging. The next application for BWR piping indicates the potential use of probabilistic SRRA results.

Intergranular stress corrosion cracking (IGSCC) of certain piping in BWR plants is an area where the SRRA methodology is being used to address uncertainties inherent in the IGSCC phenomenon. Because of these uncertainties and those associated mitigative actions, e.g., increased inspection frequency, reduction of residual welding stress with induction heating stress improvement, residual stress improvement and local resistant material additions with weld overlay, it is difficult to determine which actions would be most beneficial in reducing the risk of pipe rupture. To address these uncertainties, SRRA technology can be used to assist in developing risk-based inspection criteria for IGSCC as shown in Figure 6, which is taken from Reference (9). The PRAISEC Computer Program (10), which was developed by Lawrence Livermore National Laboratory for the NRC, was the primary tool used in the SRRA study of IGSCC. These results from the study of an actual BWR piping weld exemplify that SRRA technology can be used to provide definitive and quantitative results to assist in defining effective mitigation for the IGSCC aging mechanism, including cost-effective inspection programs which provide additional bases to manage the aging of these piping systems.

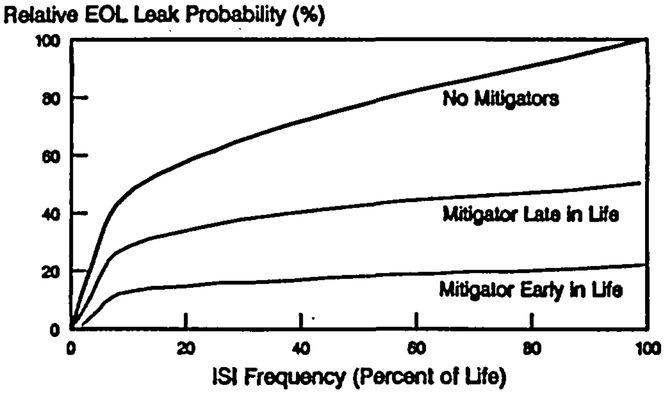


Figure 6 - Results of Initial SRRA Study of IGSCC For Use in Managing Aging in BWR Piping

REACTOR VESSEL INTERNALS

The reactor pressure vessel internals in a PWR consist of two major units. the lower and upper core support assemblies, as shown in Figure 7. The upper core support assembly provides for vertical and lateral restraint as well as lateral alignment to the top of the core. The lower core support assembly provides support to the attached internals structures and the core, load transfer to the vessel, restraint and alignment to the bottom of the core, directional and metered control of coolant flow into the core and neutron shielding for the reactor vessel. Most of the materials in the reactor internals are wrought stainless steel with some castings. Some of the smaller critical components are made of high-strength nickel alloys.

In terms of management of aging, the key degradation mechanisms were identified in a PWR life extension study (2) as: fatigue crack initiation and growth and mechanical wear, both of which are promoted by flow induced vibration

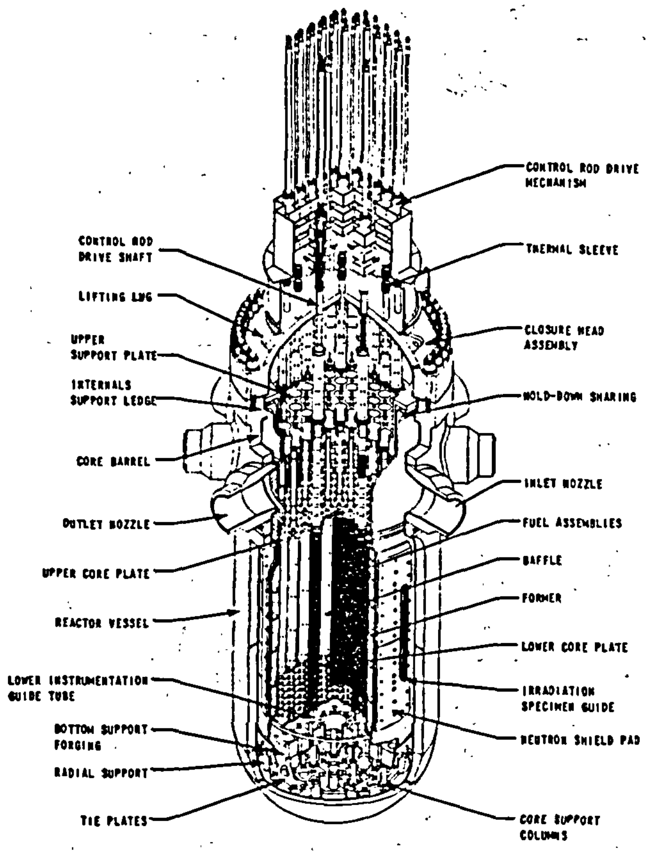


Figure 7 -- PWR Reactor Vessel Internals Components

and baffle jetting, stress corrosion cracking, and thermal aging and irradiation induced embrittlement. In each of these degradation mechanisms, there are significant uncertainties that must be addressed in any meaningful aging management program. One successfully demonstrated method of doing this would be to evaluate the inspection-repair-replace options for life extension on the basis of risk of structural failure using structural reliability and risk assessment technology. The calculated failure risks (probabilities) could then be used with failure and implementation costs to determine the most effective options or used as input to probabilistic risk assessments to determine the risk of unacceptable consequences if the component failure were to occur. Several examples of this type of application of SRRA to reactor internals components are described below.

A structural reliability and risk assessment (SRRA) of both the original and replacement support pins used in an operating PWR was recently completed. From the SRRA analysis the probabilities of failure of both pin designs were determined and compared. The purpose of the analysis was to quantify the beneficial effects of the design and material improvements that were made to the replacement pin design. The support pins, two per guide tube assembly, are the interfacing component between the guide tube assembly and the upper core plate. The support pins are bolted into the bottom flange of the guide tube and spring loaded inside the upper core plate holes to align and to provide lateral constraint to the guide tube assembly which guides the axial movement of the control rods in and out of the fuel assembly during power changes.

The results of a failure mode and effects analysis (FMEA) were used directly in the SRRA analysis to determine the critical failure modes of the support pin designs. Intergranular stress corrosion cracking (IGSCC) was determined to be the major cause of support pin failures. Some cracks had been observed to have initiated from the root of the bolt shank, and in a few cases, from the root of the leaves. The improved replacement support pin design used an improved heat treatment in order to obtain microstructures resistant to stress corrosion crack initiation and propogation. Moreover, several measures were taken to reduce the stresses in the support pins. Since these improvements also reduced the stress intensity, which controls the stress corrosion crack growth rate, they were considered important factors in the SRRA evaluations.

The support pin designs were analyzed using SRRA techniques as shown in Figure 8. A mean initial crack size that could be undetected after an initial inspection and its uncertainty was assumed based on NDE (non-destructive evaluation) experience. For the initial crack size and a randomly varying stress level in the appropriate support pin, the operating stress intensity was obtained. For the specified failure condition (normal operation or upset seismic conditions) a critical crack size was also calculated. Then the stress corrosion crack growth was calculated as a function of operating time. Failure was continually monitored and was postulated to occur when the crack depth equalled or exceeded the critical crack size. When failure occurred the effects of any prior in-service inspections were taken into account in the calculation of the failure probability, which was estimated as the ratio of failed pins to the total number of pins sampled.

This SRRA analysis showed that the probability of failure of the replacement (improved) support pin was significantly less than that of the original support pin for failure during a normal or upset event. Furthermore, the effects of a relatively imprecise in-service inspection (ISI) plan were shown to lead to even lower probabilities of failure. The overall conclusion of the SRRA analysis was that there is a high degree of confidence that the replacement pins should last considerably longer

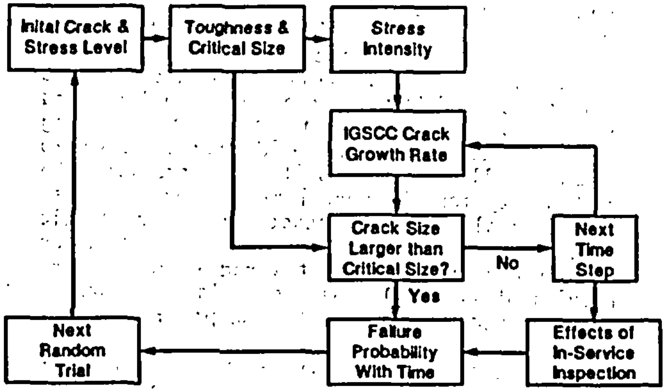


Figure 8 -- Flow Chart for Support Pin SRRA Analysis

than the original pins because of the material improvements and design changes to reduce stresses.

Another potential application of SRRA to reactor internals components would be to provide risk (probability of failure) based justification for continued operation of a PWR thermal shield and its support system to support the delay of a detailed inspection. The uncertainties that would need to be considered include those associated with the existing condition of the support system and those associated with the loading conditions during the continued operating period. This application of SRRA is especially exemplary because most of the previously defined reactor internals' degradation mechanisms need to be considered in the multiple failure modes involved in this application. This example also shows how SRRA could be used to define additional data that could be obtained during a limited inspection that would reduce uncertainty and the calculated risk if a preliminary risk analysis indicated an unacceptable result.

The risk of failure would be calculated using SRRA techniques similar to those shown in Figure 8 for the support pin but modified to specifically address the predominant failure modes and operating conditions of the thermal shield support system and their associated uncertainties. First, the stress state for the support system bolts and dowel pins would be initialized based upon the loads resulting from the thermal and flow-induced vibration analyses. Based upon this stress and the distribution of initial crack sizes, an initial crack size and corresponding stress intensity would be initialized for each trial. The changes to either crack size or stress level during steady state and transient operation would then be modelled. Mechanisms in this category could include: loss of preload, steady thermal loads, wear, stress corrosion, cracking and radiation, and installation-induced stresses. Likewise for transient changes, the effects of flow-induced vibrations, thermal transients, and earthquake cycles on fatigue crack growth could be modelled as necessary. To check for failure, the stress intensity factor for the current crack length and life-limiting transient stress would be compared to the material's current fracture toughness. If the uncertainties are such that the toughness is exceeded a failure is recorded and a new trial initiated. Otherwise the analysis with time continues forward toward end of life. A probabilistic model incorporating pertinent uncertainties would also be developed for wear at the thermal shield limiter keys. The SRRA probabilistic analyses would then be performed to generate the probability of support system bolt failure, dowel pin failure, and limiter key wear as a function of operating time. A postprocessor would then be used to combine all these results to predict the probability for the most limiting mode for loss of thermal shield function: hydroelastic instability or abnormal shield orientation or the shield falling onto the radial keys. The results of the SRRA analyses could be presented in a manner similar to the sample results of Figure 9.

Should the initial estimate of probability of thermal shield failure be judged to be unacceptably high, then sensitivity studies on the effects of the best estimate uncertainties would be performed using the available SRRA models. The objective would be to identify those uncertainties that could be reduced by additional data obtained during a limited inspection, considering the uncertainties in their measurement, and that significantly reduce the overall risk (failure probability). Once the additional data are obtained, the SRRA analyses could be repeated with the reduced uncertainties to obtain an acceptable level of risk.

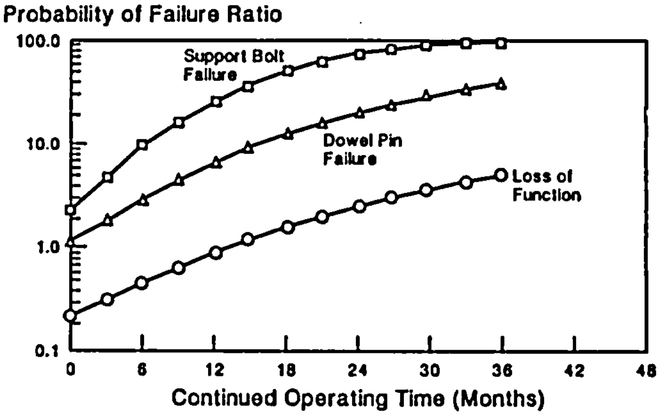


Figure 9 -- Sample SRRA Results for Thermal Shield System Operation

For SRRA analyses of reactor internals components, the general process is to first select the key component failure modes. Typically, the failure modes of concern are those with large uncertainties and significant consequences. Second, the probabilistic life-prediction models are developed, including failure criteria, key uncertainties and the beneficial effects of inspection or monitoring. Third, the risk of failure with lifetime is calculated for use in a PRA to evaluate the consequences of failure or to determine its sensitivity to the various uncertainties involved. However, its most noteworthy use is in conjunction with financial considerations, such as expected implementation and failure costs, to determine the most effective repair, replace, inspection options on the basis of risk reduction. The ultimate benefit of this approach is to provide a rational and quantitative basis for making management of aging decisions.

SUMMARY

There has been a continuing evolution of methods to predict aging and component life beginning with basic engineering principles and qualitative aging assessments and developing to sophisticated life-prediction algorithms and probabilistic methods today.

There have been significant improvements in the modeling of degradation mechanisms and the development of aging and life prediction algorithms; however, deterministic evaluations have typically resulted in conservative estimates of life. These conservative estimates of life are the result of conservative assumptions necessarily made in deterministic assessments to bound significant uncertainties.

Probabilistic tools offer an attractive alternative or supplement to the traditional deterministic aging prediction and life assessment methods. Probabilistic methods have the ability to explicitly quantify the effect of uncertainties on component life rather than conservatively bounding the uncertainties as is typically done in deterministic assessments. As a result, probabilistic methods can bring us closer to a prediction of the "real world" and enable better technical, safety, and economic decision-making. These probabilistic tools can move the assessment of risk from the hands of the analysts performing conservative life assessments to the hands of decision makers, thereby providing quantified risks associated with management of aging decisions.

REFERENCES

1. Balkey, K. R., Meyer, T. A., Witt, F. J., "Probabilistic Structural Mechanics: Chances are ....," Mechanical Engineering, Pg. 56-62 (1986) July.
2. Miselis, V.V., Massie, H. W., Beaument, P. R., "PWR Vessel and Internals Westinghouse General Approach to PLEX," ASME/IEEE Power Generation Conference, Portland, Oregon, 10/19-23/86, 86-JPGC-NE-D.
3. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.
4. Turner, R. L., Balkey, K. R., Phillips, J. H., "A Plant Specific Risk Scoping Study of Reactor Vessel Pressurized Thermal Shock," Advances in Probabilistic Fracture Mechanics, ASME, New York, Vol. 106, No. 6 87-103 (1984).
5. Witt, F. J., Bamford, W. H., Esselman, T. C., "Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," WCAP-9283, Westinghouse Electric Corporation, March 1978.
6. Lin, E. Y., "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant," NUREG/CR-2189, Vol. 9, August 1981.
7. Lo, T., et al., "Failure Probability of PWR Reactor Coolant Loop Piping," in Seismic Events Probabilistics Risk Assessment, The American Society of Mechanical Engineers, PVP-Vol. 79, pp. 11-25, 1984.
8. 10 CFR Part 50, General Design Criterion 4, "Requirements for Protection Against Dynamic Effects for Postulated Pipe Rupture," Federal Register/Volume 51, Number 70/April 11, 1985/ Rules and Regulations, pp. 12502-12505.
9. Bishop, B. A., et al., "Risk-Based Inspection For Pressure Vessels and Piping," ASME Pressure Vessel and Piping Conference, Pittsburgh, Pa., June 1988.
10. Harris, D. O., Dedhia, D. D., Eason, E. D., and Patterson, S. D. "Probabilistic Treatment of Stress Corrosion Cracking in 304 and 316NG BWR Piping Weldments," Probability of Failure in BWR Reactor Coolant Piping, NUREG/CR-4792 (UCID-20914), Volume 3, December 1986.

DEVELOPMENTS FOR IMPROVING RELIABILITY AND LIFETIME OF PWR MECHANICAL EQUIPMENT

Philippe Revel

Abstract

A survey of main lifetime technical features is presented within the framework of the French pressurized water reactor (PWR) program. Design and manufacturing evolutions, in-service surveillance, and in some cases repair methods, are presented. These features are covered within the scope of the main mechanical components of the PWR: reactor vessel, control rod drive line, pressure boundary components, and steam generators.

Introduction

Improvement of the reliability and lifetime of PWR components has been a subject of particular attention within the French PWR program due to its recognized importance (Fig. 1). This importance itself and other features, such as plant standardization and concentration of industrial tools, have created the need for large development programs which can ensure continuous progress in this area.

Development programs were launched to gain a better understanding of the degradation mechanisms on components, and to qualify new designs and materials. Initiation of these developments was due mainly to the lessons learned from the operating experience of domestic and foreign units.

Some years ago, assessing the longevity of nuclear power systems and preserving life extension potential were recognized as major concerns by nuclear plant owners. Several projects have now been implemented in this field, such as the EPRI-PLEX program.

In France, EDF has undertaken a similar project since October 1985. The first task of this "Life Duration Project" was to identify the lifetime knowledge status of the main critical components and to define complementary action plans toward improvement and completeness of this status.

These plans include: (1) the knowledge of degradation phenomena, (2) the improvement of replacement components, (3) the improvement of procedures and devices to monitor aging, and (4) the development of repair and replacement methods. The main component suppliers are associated with this project, in particular FRAMATOME as the nuclear steam supply system (NSSS) supplier.

The main NSSS critical components concerned with this project are: the reactor vessel, the control rod drive line equipment, the pressurizer, the main piping, and the steam generator. In this short presentation, we will present a survey of the main lifetime technical features for these components, including design evolution benefits along the successive PWR series, and prospects for future developments.

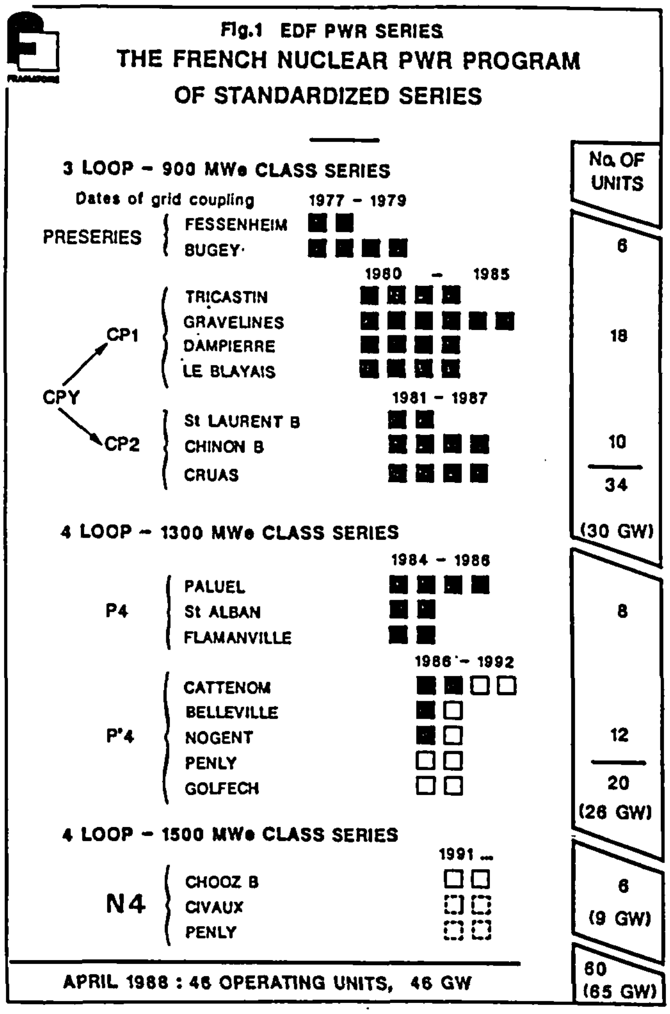


Figure 1

Main Lifetime Technical Features on PWR Components

Irradiation Embrittlement of Reactor Vessels

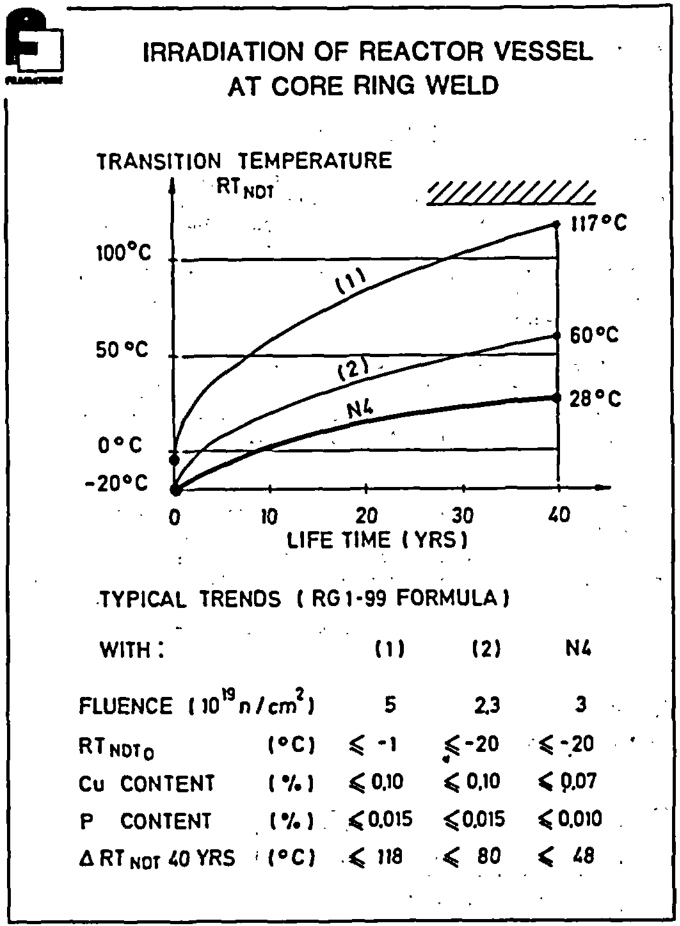
Due to its safety importance, this degradation phenomenon was the subject of a number of developments and communications. This presentation will limit itself to some design aspects.

The reactor vessel parts of concern are the core shell rings and their welded junctions. The degradation results in an increase of the transition temperature RT_{NDT} under the dependence of the integrated fluence (flux on the reactor vessel wall and operating time) and of the material properties.

For the reactor designer, several methods of alleviating this problem are possible; reduction of the flux on the vessel wall, and improvement of the material properties, are two examples. (The first is followed with the use of neutron shields installed in reactor internals.) Other possibilities include the use of a heavy reflector (considered in some advanced concepts), or the increase of the reactor vessel diameter for a given core size. This last option was judged to be not economical when evaluated for recent projects.

The main method of improvement in the French program has been the reduction of the impurity content (mainly Cu and P) and of the initial RT_{NDT} temperature of the ferritic material.

The last values specified for the N4 project and applied to all reactor vessels manufactured from 1985 were based on industrial steelmaker experience. Figure 2 gives typical trends.



WITH:		(1)	(2)	N4
FLUENCE (10^{19} n/cm^2)		5	2.3	3
RT_{NDT_0}	(°C)	⩽ -1	⩽ -20	⩽ -20
Cu CONTENT	(%)	⩽ 0.10	⩽ 0.10	⩽ 0.07
P CONTENT	(%)	⩽ 0.015	⩽ 0.015	⩽ 0.010
ΔRT_{NDT} 40 YRS	(°C)	⩽ 118	⩽ 80	⩽ 48

Figure 2

Control Rod Drive Line Equipment

The EDF requirements for high plant maneuverability performances (frequency control, load follow, etc.) resulted in an increased duty for the control rod drive line components of up to 3×10^5 steps per year, compared to the design life value of 2.5×10^6. The sensitive parts are the control rod cladding, which is submitted to wear by fretting in the guidance structures, and the grippers of the latch assembly, which are submitted to fatigue, abrasion, and wear.

Long-run endurance tests at full scale and full operating conditions were performed at the SUPERBEC facility at Cadarache research center in order to evaluate the behavior of the reference equipment and the effectiveness of different methods of improvements. These are some of the tests that were performed:

- Application of coatings to reduce wear (e.g., tungsten-carbide on grippers, chromium-carbide on control rod cladding, etc.).

- Hydraulic modification of the inner shape of the rod guides [3 loop-900 MW(e) reactors], then provision for additional flow holes in the lower part of the rod guides, thus improving pressure distribution and reducing friction loads [4 loop-1300 MW(e) reactors].

- Replacement of single-tip grippers by double-tip grippers [4 loop-1300 MW(e) reactors].

The endurance tests produced these results:

- Wear can be significantly reduced by coating applications.

- Wear and fatigue are more severe on 1300 MW(e) drive line components than on 900 MW(e) ones.

- The test with the reference guide tubes resulted in control rod cladding failure after 1.5×10^6 steps, while hydraulic improvements limited the control rod wear below the allowable limit for 15-year service life (Fig. 3).

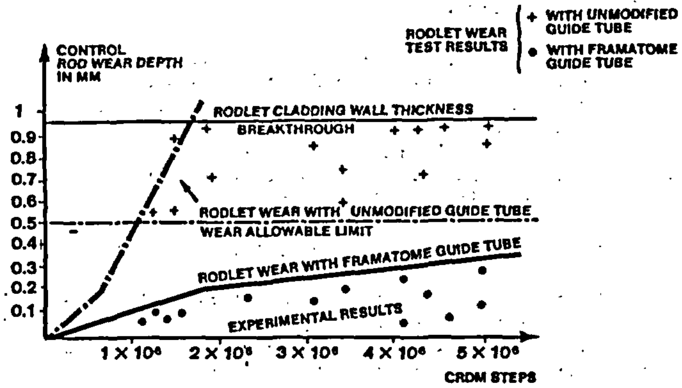


Figure 3

- Fatigue failure of a single-tip gripper was observed after 6×10^6 steps, while the importance of abrasion products and wear was reduced with the double-tip gripper (Fig. 4).

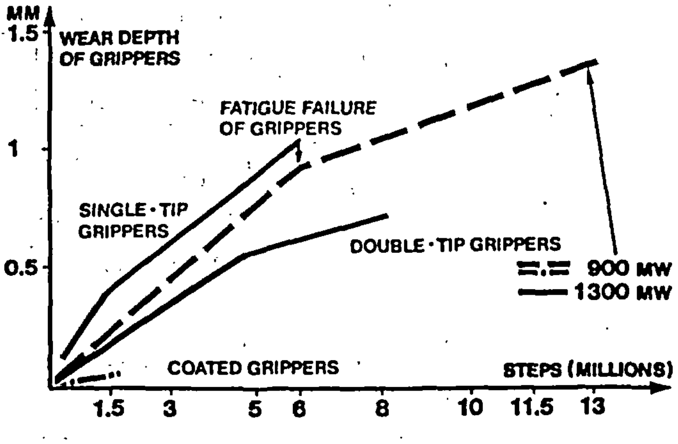


Figure 4

These results, issued in the early 1980s, contributed to the award of the load-follow operating license for the 900 MW(e) reactors in May 1984.

Forecasts for future developments related to the lifetime of these components include:

- An in-service surveillance system based on analysis of noise and electric signals.

- The optimization of the mechanism operating cycle to reduce the shock intensity between the grippers and the drive shaft.

Thermal/Mechanical Fatigue of the Reactor Coolant System Components

This degradation mode is a more or less severe concern for all the components which belong to the pressure boundary of the reactor coolant system, and for associated under-pressure systems. They are submitted to pressure and temperature transients related to operating conditions.

Several mechanisms of fatigue may occur:

- Fatigue in current parts of components (vessel shell, piping, etc.).

- Initiation and propagation of cracks in local areas where geometrical profiles may generate stress or strain concentration.

- Growth of flaws.

Fatigue in current parts is generally not a concern. Permanent temperature fluctuations have been observed in some pipes in low-flow situations (like the pressurizer surge line) where natural convection/stratification of the fluid may be predominant. Onsite temperature measurements performed at the CRUAS plant confirmed this phenomenon. This problem is alleviated by avoiding horizontal branch connections as much as possible, and by pipe routings.

At the request of the French Safety Authorities, the plant operating conditions must be monitored and documented by the plant owner, and then compared to the allowance given in the design transient file. Tools and procedures have been set up by EDF and FRAMATOME to perform this monitoring, and they have led to the design of an operating transient monitoring system featuring transient recording and real-time damage evaluation.

Thermal Aging of Austeno-Ferritic Steels

This problem mainly concerns the cast steel elbows of the main loop piping. This aging, due to a thermal degradation of the metallurgical structure above 280°C, results in a decrease of mechanical properties (toughness, impact strength, etc.). This problem is more critical for steel with molybdene content. On the basis of the present knowledge, steels without molybdene would have acceptable mechanical characteristics after 40 years of service.

Developments presently underway in this area apply to:

- Devising appropriate control procedures to evaluate the actual embrittlement of in-service components.

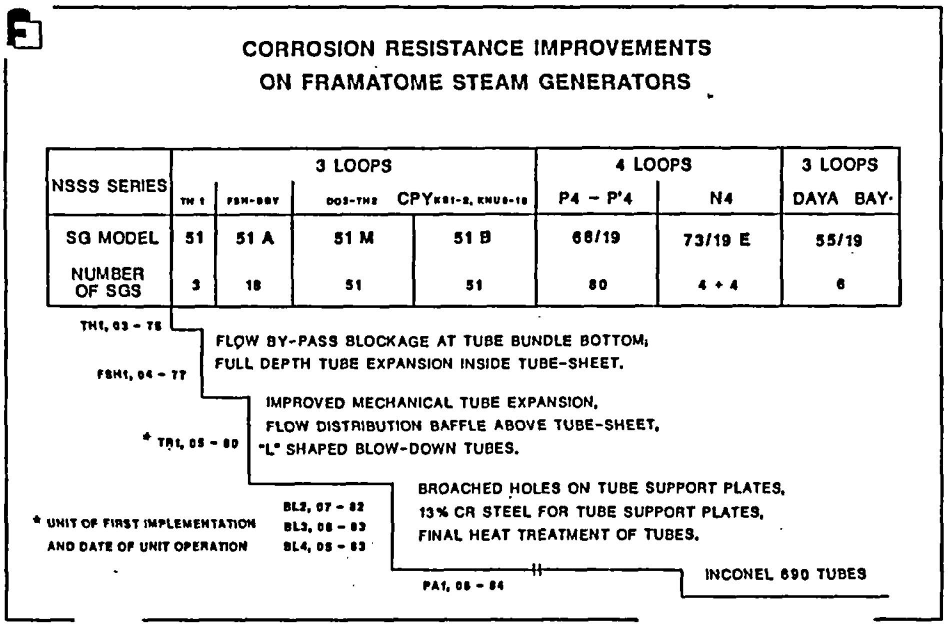
- Manufacturing alternatives to elbows.

Reliability of the Steam Generators

It is well known that the steam generator is one of the most sensitive components due to the numerous degradation mechanisms to which the tube bundle can be submitted: tube thinning (or wastage), stress-corrosion cracking on primary or secondary tube side, intergranular attack (IGA), and denting, pitting, and fretting wear.

Sensitive parts are the tube itself, the vicinity of the secondary side of the tube sheet, the tube to tube-sheet junction, the tube to tube-support interface, and the small radius U-bends.

Based on the operating experience of foreign units, a continuous effort in the areas of design, material selection, and the manufacturing process has been pursued; the overall goal is to improve the reliability, and especially the tube bundle resistance, of the steam generators. The historical implementation of these improvement features is given in Fig. 5.



NSSS SERIES	3 LOOPS				4 LOOPS		3 LOOPS
	TH 1	FSH-BBY	DOS-TH2	CPY KB1-2, KNU5-10	P4 – P'4	N4	DAYA BAY
SG MODEL	51	51 A	51 M	51 B	68/19	73/19 E	55/19
NUMBER OF SGS	3	18	51	51	80	4 + 4	6

Figure 5

Design improvements were aimed at increasing the recirculation ratio and the flow velocity above the tube sheet, increasing the subcooling at the hot leg inlet, extending the blowdown suction in the low velocity areas, and improving the tube scavenging at the tube supports. These improvements were accomplished by the elimination of flow resistance and bypass in the recirculation loop, and by the use of flow distributors, appropriate feedwater distribution, L-shaped blowdown tubes, and multifoiled tube holes on tube support plates.

Material improvements include the implementation of a final thermal treatment of the Inconel 600 tubes (720°C for 15 hours in a vacuum environment), which eliminates the residual stresses and improves the metallurgical structure. More recently, after completion of a large research and development program from 1976 to 1985 (Fig. 6), a new alloy, the Inconel 690 (30 Ni FeCr), was chosen as the best available material; it was implemented at an industrial stage on FRAMATOME steam generators. Another material improvement was the use of a 13% chromium steel for the tube support plates.

Manufacturing improvements related to the tube resistance include the integral mechanical tube expansion, whose functions and parameters are performed by an automatic machine under computer control, thus insuring a high degree of process reliability and repetitivity. A new "row-by-row" tubing process has also been implemented recently. It allows a better control of the gap between the tubes and the anti-vibration bars, and of the geometry in the U-bend area.

All of these improvements, progressively introduced into the steam generator series, are now available for steam generators of recent units as well as for replacement steam generators. The replacement steam generator design also takes into account provisions for heat transfer surface margins, which give a higher assurance of full-power lifetime.

In addition to the above-mentioned improvements in steam generator design and construction, a number of in-service surveillance methods, preventive operating procedures, and repair processes have been developed to control and extend the lifetime of steam generators. Water chemistry modifications and the hideout process are examples of preventive operating procedures. Available repair processes now include tube plugging or sleeving, shot peening or roto peening, thermal treatment of small U-bends, and anti-vibration bar replacement. Further developments in these areas are underway.

Conclusion

This short survey of lifetime concerns for the main reactor components shows that a significant amount of knowledge and industrial experience has been acquired in this area. Modern units must present a higher resistance to degradation phenomena than did the previous units. Refurbishment methods and new design features now exist that enable us to restore or replace the more sensitive components which are susceptible to operating problems over time.

Further developments are still required to anticipate potential problems and to improve in-service surveillance. The goal is to supply plant owners with better tools for future optimized component lifetime management.

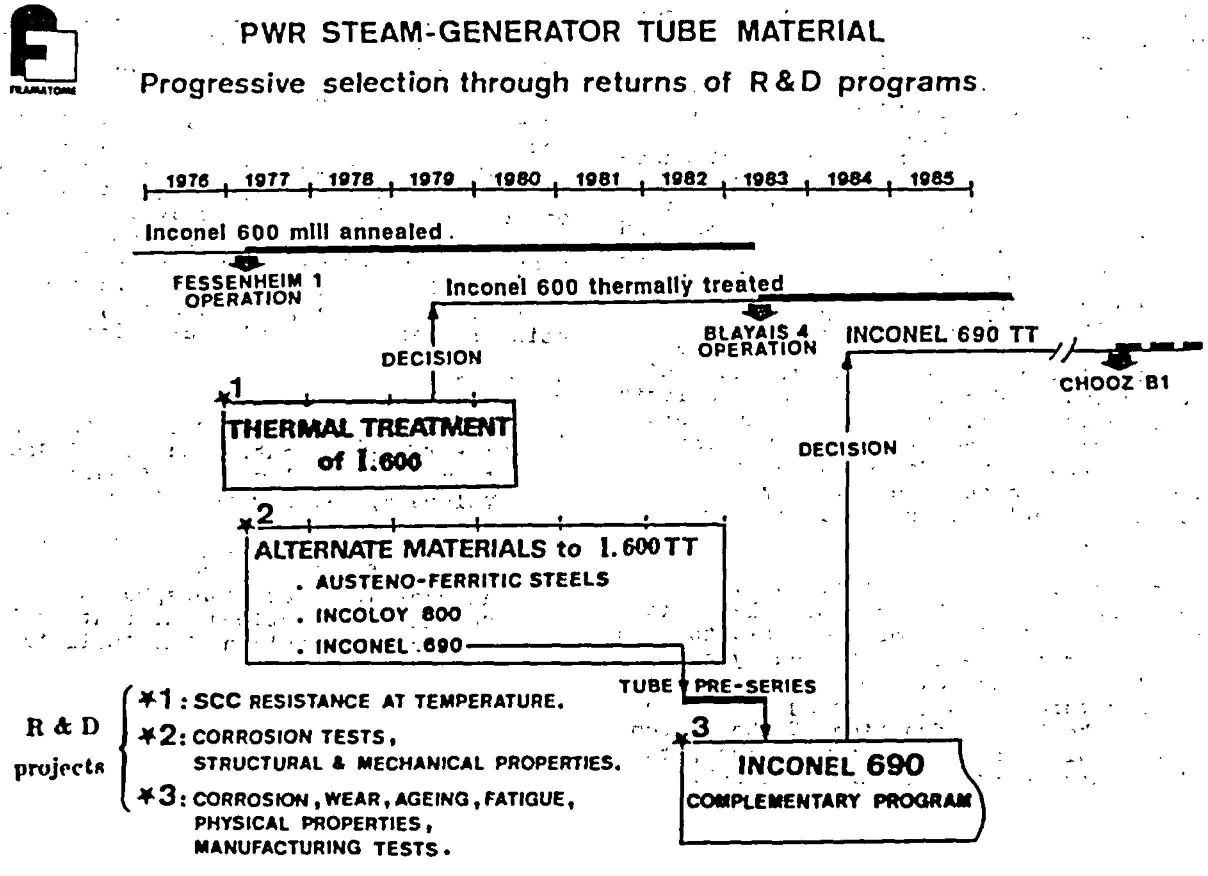


Figure 6

RELIABILITY PROGRAMS: A TOOL FOR MANAGING AGING RISKS

Joseph R. Fragola
John Wreathall
Science Applications International Corporation

As the early generation of nuclear power plants approach their nominal design life time, there is growing concern about the levels of risk to public safety. NRC has, for example, implemented a multiyear research program to investigate what influence aging may have on risk, and which are the most important components from a risk-management perspective.

Initial emphasis has been largely given to an evaluation of components for which reliability performance data can be collected, to examine whether there is, in fact, any trend exhibited for an increasing failure rate with time. To date, results have been mixed; some analyses appear to indicate an increasing failure rate and some do not. One reason for this lack of clear evidence is believed to be changes in the quality of maintenance data from the early days of nuclear plants to now. The records associated with early plant operations may not be sufficient to identify the causes of failure or the extent of repair; nor is it often easy to estimate the number of successful operations from the early days.

However, it is possible that another mechanism is at work to prevent increasing failure rates with age, the adoption (either consciously or fortuitously) of maintenance practices that control the degradation of components from aging. One such approach being given extensive consideration in the nuclear industry is reliability-centered maintenance. In such an approach, maintenance practices are adjusted to preserve the reliability of significant components within target bounds. For example, the frequency and extent of preventative maintenance actions can be adjusted based on experience; condition monitoring can be applied to specific kinds of components; the feedback of root-cause analyses can lead to change in maintenance activities. All of these, which are part of reliability-centered maintenance, act to limit the increase in unavailability of components like pumps, valves, and diesels. Therefore those plants that have adopted such an approach to maintenance are unlikely to exhibit any effective increase in failure probabilities.

This limitation on the increase in failure probabilities at the system level is predicted by Drenick's Theorem.[1] This theorem states that for any series-oriented system having a large mixture of components with a mixture of hazard rates (including some time dependent), the hazard function of the system is bounded. With repair and replacement, the system's hazard bound is lower and reached quicker.

Given, then, that systems having the characteristics of Drenick's postulate (almost all nuclear plants safety systems) will only exhibit a limited increase in hazard rate, other components may be more significant from the aging-related risk. These will include components not subject to monitoring and repair that are not substantially large series-oriented. These included the large structures (vessel supports, containment, etc.) and passive components (piping, tanks, cables, etc.)

These are components not normally represented in PRAs because of their normally low failure rates, and they are not normally included in reliability monitoring. However the consequences of their failure may be very great.

The difficulty with incorporating these structures and components into a reliability-centered maintenance program is principally one of lacking any means of "testing" them in any meaningful way. NDT can be applied to piping and tanks; overpressure testing can be applied to containment to a limited degree; yet these are not considered complete and effective. Until such methods can be developed, for those plants having effective reliability programs, we believe that structures and passive components pose the greater risk. We believe that the Surry feedline break and (in aviation) the Aloha jet crash are both events that support this theory.

SUMMARY

It is believed that with proper attention to maintenance management aging effects on active repairable components can be limited in their impact on safety to a significant extent. A reliability program, such as that described in NUREG/CR-5078,[2] is an example of an approach to maintenance management that can effectively limit aging. The effect of such a program will be to limit unavailabilities within an acceptable range, though with an increased frequency and downtime for repair--ultimately the unavailability will be dominated by the repair downtime, at which point replacement should take place.

REFERENCES

1. Drenick, R.F., "The Failure Law of Complex Equipment," Journal of Society of Industrial Application of Mathematics, Volume 8, #4, December 1960.

2. E.V. Lofgren, et al., A Reliability Program for Emergency Diesel Generators at Nuclear Power Plants, NUREG/CR-5078, Science Applications International Corporation, Mclean, VA, April 1988.

DEVELOPMENTS IN EDF POLICY WITH REGARD TO MONITORING THE AGING FACTOR IN PWR NSSS

G. Bimont
G. Cordier
Electricité de France

1. INTRODUCTION

When the first 900 MWe PWR started operating, in order to satisfy French regulations, EDF set up a relatively simple procedure to monitor the transients affecting the PWR main primary system. This procedure, TMBP (Transient Monitoring and Bookkeeping Procedure), which will be outlined in this paper, is aimed at confirming the conservativeness of the design hypotheses.

It also provides a system of chronological data on operating loads, which is of considerable interest from the point of view of improving nuclear steam supply system (NSSS) maintenance, optimizing the operation, and extending lifetime beyond the value initially anticipated.

However, EDF rapidly became aware of the limits of this procedure due to its simplicity. On this basis, EDF has been developing for several years on-line monitoring systems of the usage factor in special areas of the main primary system. The experience performed with a fatiguemeter prototype (operating at Bugey 2 power plant since 1985 and then introduced at Dampierre 1) has shown interest to develop this kind of system, as a complement to the present procedure, with a view to monitoring a limited number of highly stressed areas where a close assessment of fatigue aging is vital.

EDF has consequently decided to gradually equip the PWR plant units with monitoring networks, which will be briefly described. These devices will be designed and realized in collaboration with FRAMATOME, manufacturer of the PWR NSSS.

2. EDF'S EXPERIENCE OF LOADING MONITORING IN PWR NSSS

As we have previously mentioned, EDF has implemented, since the startup of the first 900 MW PWR plant unit, a procedure to verify that the actual transients to which the main primary system is subjected are less severe and less frequent than those taken into account by the design hypotheses. This verification is based on the comparison of the actual temperature and pressure changes with those anticipated during the design phase for various situations. The consequences, in terms of stresses or damage, are not the subject of any direct assessment. While at the beginning of the project the possibility of on-line stress computation was already considered, the TMBP procedure, simply and rapidly implemented, is still in use.

The procedure has already been described in several publications[1-3] in which the reader may find a more detailed description. The procedure, which is based on normal operating recordings of the main physical parameters representative of the plant solicitations, was developed:

- To detect any variation in the parameters capable of creating significant stress variations and to thus increase the equipment usage factor (i.e., the fatigue "damage"),
- To identify the functional origin of these variations, by relating them to what is known as an operating transient,
- To assign this transient to an envelope design transient in the design list,
- To count the design transients realized and make sure that the predicted numbers of occurrences determined during the design phase are not reached.

This procedure therefore attempts to ensure that the damage from equipment fatigue is not greater than that foreseen by the design hypotheses. Verification of the other types of damage (excessive deformation, elastoplastic or plastic instability) is obtained by checking that the design pressures and temperatures are not exceeded.

2.1 Design Transients List

The NSSS's behavior was analyzed on the basis of this published procedure, which is in two parts:

- The list and description of the design transients with their estimated number of occurrences for the entire life of the NSSS,
- The NSSS's main physical parameters' variation curves (primary and secondary pressure loop temperatures, ...) as provided by the computation models (eventually adjusted to test results), which characterize each design transient.

These design transients (a limited number, approximately 30 of them concerning the reactor coolant system (RCS), and 20 specific to other auxiliary circuits), were designed so as to:

- Represent and envelop a certain number of actual elementary design transients (for example: the "50% load variation" design transient covers any lesser change in amplitude),
- Take into account, with respect to the most probable actual operating transients, a certain number of conservative hypotheses, which generally are representative of a succession of equipment or system failures. (For example: the "turbine trip with partial opening of the turbine bypass" design transient takes account of the unavailability of a pressurizer relief valve.)

This procedure was developed by the manufacturer, in collaboration with EDF, based on experience and feedback from operating plants, and in consideration of the operating mode intended for the plant. It serves the operator as a basis for transient bookkeeping.

2.2 Detection, Identification, and Assignment: Three Essential Phases of Bookkeeping

The detection phase is due to the impossibility of considering all fluctuations of the parameters. Small fluctuations were eliminated from the procedure by determining parameter thresholds below which the structure is shown to undergo no fatigue damage. These thresholds were estimated by performing thermoelastic analysis on the most sensitive and significant regions of the primary circuit. These studies led to the presently adopted threshold values:

primary pressure: 10 bar
hot leg temperature: 5°C
cold leg temperature: 5°C, etc.

Furthermore, some operating events are monitored since they are significant with respect to the fatigue aging of several particular areas (for example: opening of a pressurizer relief valve).

In practice, as soon as a threshold value is exceeded, or a particular event has arisen, a "significant operating transient" is considered to occur. This happens once or twice a day per plant unit. Without going into details, we will add that it was necessary to define other guidelines to determine the end of the transient, separate "neighboring" transients with respect to time, and take eventual "subcycles" into account (i.e., small, superimposed transients), etc.

Identification consists of locating the functional origin of the transient whenever a significant fluctuation in temperature or pressure is observed on the recorders. This stage is essential for several reasons:

- There are cases in which physical measurements, because of the position of the instrumentation, do not enable an accurate assessment of the actual severity of the transient (in the case of isolation of a system, for example),
- Direct checking, in the design transients file, for an envelope transient curve of the variations of several parameters is difficult and may lead to error (especially with regard to the assessment of temperature gradients),
- Lastly, characterizing the transient enables, a posteriori, analysis of the causes of malfunction (transients too severe or too numerous) and the provision of solutions.
- The identification is made according to operational criteria (nuclear power, state of various actuators, turning equipment on or off). For this, the operator used the log book written by the operation teams and data supplied by the plant unit computers.

Once the transient has been identified, the last step consists of assignment. One or more transients are located, in the design transient list, that are at least as severe in terms of equipment fatigue. Simple criteria were selected (parameter range, rate of temperature change, number of subcycles) to do this assignment.

In practice, there are three possible cases:

- Transients that are identifiable via a functional approach: They are those that, during the previous step, could be simply linked to a perfectly defined operating event. Then the validity of the associated design transient must be checked.
- Nonidentifiable transients: Assignment is made, without functional association, by checking for the transient (in the design transient list) whose changes in temperature and pressure most nearly envelop the actual changes (to avoid a too pessimistic aging assessment).
- Nonassignable transients: The two previous methods have failed. These transients are the subject of a particular analysis. If such transients offer no general characteristics, simple thermomechanical analyses are used to find a representative (conservative) design transient. On the other hand, it may be due to an oversight in the design transient list or unexpected parameter changes.. The possible solutions are then that either the operator modifies the operating conditions so as to prevent such severe transients from recurring, or the operator may work with the manufacturer to modify the Design Transient List and correct the design analyses. Of course, the integrity of the Reactor Coolant Pressure Boundary (RCPB) must be checked with these new design hypotheses.

2.3 Experience Feedback

Experience feedback shows:

- Good conformity between the operating transients and the design transients with respect to parameter variations,
- In general, more severe design transients,
- Fewer transients than estimated.

We will now illustrate these affirmations using a few examples.

Analysis of the "off-site power loss" transients which occurred in PWR 900 power units shows that in 154 reactor years, 67 transients were recorded as "off-site power loss." It should be noted that this does not mean that there actually were 67, but simply that 67 operating transients had temperature and pressure changes represented conservatively by this design transient. Furthermore, out of these 67 cases, 49 were recorded before commercial operation. We also notice that the design hypothesis was one transient per year, therefore 154 for the same cumulated operating period. Furthermore, out of those 67 actual transients, only 8 led to exceeding the SG valve opening threshold levels, contrary to the design transient hypothesis.

For 1 of these 8 cases, Figures 1 and 2 show the comparison between actual temperature and pressure changes and design values. The less severe nature of the physical parameter changes during the operating transients clearly appears.

Another example involves load changes. Figure 3 provides the number of "large load changes" recorded for 4 plant units. It may be observed that we are far below predicted results, but except for FESSENHEIM 2, the tendency is toward a very clear increase in load following (which may be explained by the increasing number of nuclear plants operating in the EDF grid).

In fact, with regard to the NSSS general transients, the main modifications in the list of design transients consisted of including the new design transients, which were the result of new operating modes (stretch-out operation, power plant uprating) or equipment modifications (modification of RCS overpressure protection system).

The auxiliary systems, subject to more complex phenomena, were the source of more problems, the greatest of which was the charging line of the chemical and volume control system (CVCS).

The transients of this line and of its nozzle on the primary loop are carefully monitored since the line and nozzle have a high design usage factor. Furthermore, the first 900 MW plant units have a welded thermal sleeve on the nozzle, the resistance of which was a source of trouble.

This monitoring quickly made it very clear that, in spite of the reasonable overall numbers of transients recorded (Figure 4), there was significant overconsumption of design transient n° 37 (isolation of the CVCS letdown line). (In Figure 5, the number estimated at the end of 8.5 years of operation is compared with what was observed at BUGEY 2.)

In terms of usage factor (in the sleeve-piping welding) and considering the underconsumption of other design transients, there was no trouble with this (Figure 6), but it still led us to do the following:

- Make the operators more aware of this, so that they improve plant unit operation,
- Modify pressurizer level control to limit thermal shock on the line,
- Correct the design analyses using a more realistic assumption.

2.4 EDF's Interest in Monitoring

From what has just been said, it may be seen that transient bookkeeping provides important results concerning plant unit operation which may be used as aging indicators.

These indicators may be used for the following:

- Conditional maintenance,
- Optimization of plant population management to obtain homogeneous aging of all the plants,
- Justification of a request to obtain plant life extension made to the Safety Authorities.[4]

Nevertheless, the limitations on the current procedure for transient bookkeeping must be taken into consideration:

- All the areas of the reactor coolant pressure boundary (RCPB) are not subject to monitoring by transient bookkeeping. The surge line is subject to phenomena that are too complex (stratification, back and forth water motion) and instrumentation that is too far away from the critical locations for the method to be applicable. However, it is an area that calls for close fatigue monitoring (complex geometry, sensitivity to small fluctuations in temperature, high design usage factor, etc.).
- Transient bookkeeping fulfills correctly the monitoring required by French regulations. However, it does not provide an accurate idea of actual aging. In particular, this procedure does not take into account the slightest severity of the actual transients in comparison with design values that would be useful for plant life extension.
- As we have seen in the case of loss of off-site power, there is no perfect correspondence between operating events and recorded transients, hence a certain difficulty in interpreting, for operating feedback, the transient bookkeeping results.

These three essential reasons (the first two being most important) very early led EDF to develop other monitoring methods that are more accurate and thus more revealing, as we will now see.

3. FATIGUEMETER EXPERIMENT IN EDF'S PLANTS

3.1 A New Monitoring Concept

As said before, as soon as design transient bookkeeping was implemented, EDF was considering new ways of meeting its statutory requirements regarding NSSS transient monitoring. Thought was focused on two objectives.

1. Automation of this monitoring (TMBP), which was judged to be difficult and delicate to carry out.
2. Improvement in monitoring quality by verifying the conservative value of the design hypothesis, no longer using the temperature or pressure changes but using the stress changes, or preferably, using an assessment of the resulting damage (usage factor).

To do this, we have selected the following method: development of a new system enabling real time access to local stresses and to the resulting damage using only existing operating measurements.

These ideas were only realizable via 1) development of simplified mechanical computation methods (transfer functions) that were compatible with on-line monitoring, and 2) the accumulation of sufficient knowledge regarding the behavior of the most critical areas of the RCPB. The local thermohydraulic conditions (flow rate, temperature) may thus be accurately assessed using only available operating measurements.

These computation methods were developed and used at the beginning of 1980 to analyze test data.[5] The knowledge required to establish models in the critical areas was obtained using local on-site instrumentation; the following may be mentioned:

- Charging line nozzle - Bugey 5 in 1980,
- Surge line nozzles - Cruas 2 in 1985.

In 1984, a prototype monitoring system known as the fatiguemeter was developed.[6] In the beginning, it was designed to supplement the bookkeeping system and enable the fatigue aging of certain critical points of the RCPB to be assessed more realistically.

3.2 Operating Principle

Data processing of the operating measurements enables assessment of the damage in the most sensitive areas of the RCPB.

This software, loaded into a modular type minicomputer, performs several functions (Figure 7):

- Acquisition and checking of the operating parameters required to monitor the studied locations in a period of 10 seconds,
- Computation of local boundary conditions: This is an important function which determines the validity of the overall system. It uses EDF's feedback (on-site instrumentation, test loops, mockup) in the determination of experimental correlation between the operating parameters and local boundary conditions.

  This solution (using only normal operating instrumentation) is a benefit of plant series standardization, by using the correlations validated on a plant unit for all of the identical plant units. Figures 8 and 9 show the validation of these modules for several locations.
- Stress computation: This computation is made continuously as new physical values processed by the previous module are acquired.

  This was made possible by the use of simplified methods, in particular of transfer functions for the calculation of thermal stresses.[5] These transfer functions are determined in advance by elastic analysis, using finite elements, of the response to a typical transient (temperature linear variation).
- Detection and classification of the operating transients according to their severity, i.e., their maximum stress variation.
- Storage of peak and valley stress values on a high-capacity mass storage system after prior filtering.
- Computation of usage factor, using an algorithm which takes into account the transient chronology (counting the cycles using a "Rain Flow" method and linear cumulation of the damage).
- Printing and displaying main data: Change in transient, stresses, and usage factor, cumulative and periodic.

3.3 Operating Fatiguemeters

At the beginning of 1984, an experimental fatiguemeter was installed in a 900 MW PWR operating at Bugey 2. This first device only monitored one area: the charging line nozzle on the cold leg of the primary system. This area was selected because of the number of transients to which it is subjected, which led to a high usage factor; it furthermore benefits from numerous experimental results acquired by Electricité de France during on-site testing programmes, thus facilitating modeling of the local boundary conditions.

At the beginning of 1988, a new fatiguemeter was installed at the Dampierre 1 plant (900 MW PWR). This device is installed on a minicomputer with a UNIX multitask management system enabling work on several areas of the primary system at the same time.

This system detects and classifies the surge line transients and computes the usage factor for two areas: the surge line pressurizer nozzle and the hot leg surge line nozzle. These two areas are subject to severe loadings, especially during the average temperature fluctuations caused by load variations due to frequency control. This system is currently being extended, and monitoring of the surge line, residual heat removal injection nozzle, and steam generator feedwater nozzle will be added this year. The feedback from this type of device will be used to define the future NSSS fatigue monitoring system presented below.

3.4 Results

The results are essentially those provided by the Bugey 2 fatiguemeter; the Dampierre 1 fatiguemeter is too new to have a complete feedback.

However, in spite of the short experimentation period of the "surge line" Dampierre 1 fatiguemeter, first put into operation at the beginning of 1988, it now seems that the actual transients are less numerous than expected. This is largely due to the plant operator's feedback (modification of the pressure control operation conditions enabling improved thermal behavior of the surge line).

3.4.1 Automatic Transient Bookkeeping

This is carried out according to the actual severity of the transients taking into account the stress variations, and not according to a "curve comparison" of design and operating transients.

More generally, by taking more accurate account of the transient influence (slope influence), the fatiguemeter proves the excessive severity of current TMBP.

This is clearly seen in Figure 10 where fatiguemeter transient bookkeeping and current bookkeeping are compared.

A general shifting of the most severe transient to the less severe transients may be noted. Furthermore, the most penalizing design transient in terms of lifetime was not recorded by the fatiguemeter, although it was recorded very conservatively by the plant operator.

3.4.2 Usage Factor

From what has been said, we may expect a usage factor much less than expected during the design phase, and even much less than that estimated from the results of traditional bookkeeping.

This may be observed in Figure 11 where the change in usage factor over two years of operation is indicated.

It must be noted that since the system was installed well after the plant unit was put into operation, the usage factor is initialized using the consumption of already obtained transients.

An excellent correlation between the usage factor and the most severe transients may also be observed in Figure 12 (letdown line shut-off, orifice closure on the letdown line).

Figures 13 and 14 indicate the stress spectrum and the resulting usage factor over a year of operation.

In conclusion, the experiment begun in 1985 on Bugey 2 allows us to hope that a lifetime much greater than the design lifetime may be obtained, in spite of the high frequency of certain transients. However, care must be taken when making this extrapolation, since modifications in operating modes in the years to come could alter these results, and it would be wise to constantly monitor this area.

Considering these first results, this prototype system will gradually ensure monitoring of other sensitive areas of the RCPB: the charging line (same module as Bugey 2), the cold leg RHR nozzle, and the feedwater nozzle.

Extending to a 1,300 MW plant unit is also being considered.

4. CHANGES IN EDF'S POLICY REGARDING FATIGUE MONITORING OF THE PWR NSSS

After ten years of practical experience in transient bookkeeping and three years of experimentation on the fatiguemeter, EDF believes that it is time to consider the future of PWR NSSS fatigue aging monitoring, using the experience gained in these areas.

The launching of the PWR Life Evaluation Project played a large role in this decision; stress monitoring is essential in proving that the operating condition of the NSSS allows extending its commercial operating period.

A work group including representatives of the three divisions concerned (Production, Equipment, Research and Development) was given the task of finding a monitoring system that would meet the needs of EDF, be more effective than the current design transient bookkeeping procedure, and be automated instead of manual.

It should be noted that this work group, although internal to EDF, was in contact with the NSSS manufacturer, FRAMATOME. A future collaboration, based upon technical exchanges between EDF fatiguemeter knowledge[7] and FRAMATOME work on fatigue monitoring[8] will be considered.

The monitoring system proposed would have a certain number of options resulting from fatigue monitoring feedback:

1. The system is supplied with operating data, no additional instrumentation is required. This choice, which means simplicity and cost reduction, is made possible via the standardized plant effect (cf. paragraph 3).

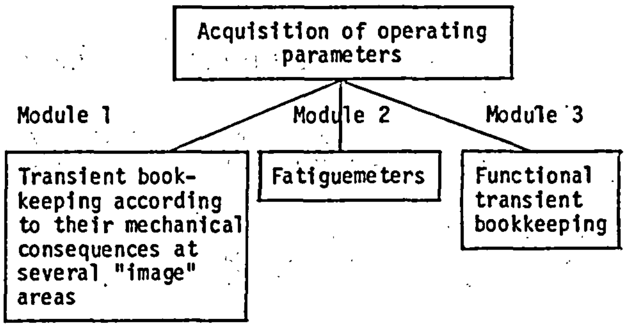
2. The sophistication of the methods to be used to monitor various points of the RCPB must be adapted to the extent of risk and the design margins that are sought. Then, fatiguemeter type methods are to be reserved for a few very stressed areas (surge line, charging line, and the RHR injection line) which are subject to complex phenomena that may not accurately be monitored by a transient bookkeeping method.

3. Transient bookkeeping according to the mechanical consequences of the transients with respect to the NSSS must be associated with the recording of the operating incidents that occur.

   This is indispensable in order to be able to interpret, a posteriori, any drift observed with respect to either the computed usage factors or the number of transients recorded, and to then propose solutions.

4. The system must be on-line with data acquisition and fully automatic.

From these choices, the system chart is the following:



The purpose of the first module is to roughly monitor the fatigue in the overall NSSS. This will be obtained by selecting a few zones representative of the NSSS behavior. Then the transient bookkeeping will be carried out, contrary to current procedure, no longer according to changes in T and P, but using a very simplified computation of the stress response of these zones to an operating transient.

This response will be compared to the precomputed values associated with the various design transients. The actual transient will be assigned to the design transient whose severity is just greater than its own. We therefore see that the general methodology of the transient bookkeeping procedure will be maintained.

For the most critical locations (high design usage factor, transient bookkeeping feedback, manufacturing and in-service inspection results), a more precise computational method via transfer function will be used and a real usage factor computation will be made (module 2). This will offer the advantage of a better determination of the fatigue due to the transient, taking account of its actual shape, and no longer be assimilating it to a design transient which may be very enveloping. The methods used will be those described in paragraph 3.

The objective of module 3 will be to record the actual operating events (reactor trip, load rejection, load changes, etc.) which occur along with their dates. This will be done using simple operating criteria (logic signals). The events will be identified from a list of operating transients which could be much more complete than the list of design transients used in module 1.

In practice, these systems will be installed in a minicomputer and connected to the on-site computer system, so as to ensure management of monitoring data and results on the local and national levels.

5. CONCLUSION

Fatigue monitoring of the NSSS is an important factor in optimizing PWR maintenance and operation. It is also indispensable for proving that the NSSS may last longer than initially expected. Aware of this challenge, EDF has undertaken an ambitious automatic monitoring system development programme which is based on the large amount of experience gained over the last 10 years, using the transient bookkeeping procedure and experiments on prototype fatiguemeters.

REFERENCES

1. R. Noel and J.P. Mercier. Bookkeeping the operating transient in EDF PWR plants. SMIRT-6 Conference, Paris, 1981.
2. J.P. Hutin. Reactor pressure vessel integrity and inservice inspection. SMIRT-7 Conference, 1983.
3. G. Cordier & J.P. Hutin. Suivi en exploitation des transitoires subis par les chaudieres REP. Comparaison aux situations définies à la conception (Monitoring during operation of transients undergone by pressurized water reactors (PWRs). Comparison to conditions defined on design.) Journées d'études AFIAP, Paris, October 1986.
4. R. Noel. PWR Life Evaluation Project at EDF. SMIRT-9 Conference, 1987.
5. G. Bimont. Comparison between real and design thermal transients in the charging line nozzle of a pressurized water reactor. SMIRT-6 Conference, 1981.
6. G. Bimont. Suivi en temps réel des contraintes subies par une zone particulièrement sollicitée du circuit primaire lors de variations de charge. Réunion UNIPEDE, Paris, 1985.
7. G. Bimont & P. Aufort. Fatigue monitoring in nuclear power plant. SMIRT-9 Conference, 1987.
8. B. Boneh et al. (FRAMATOME). The use of data processing applied to operating transients affecting auxiliary system piping and components of PRW. SMIRT-9 Conference, 1987.

FIGURE 1 : EVOLUTION OF PRIMARY SYSTEM TEMPERATURE DURING A "LOSS OF OFF-SITE POWER" AT BLAYAIS 4 - COMPARISON WITH DESIGN HYPOTHESIS

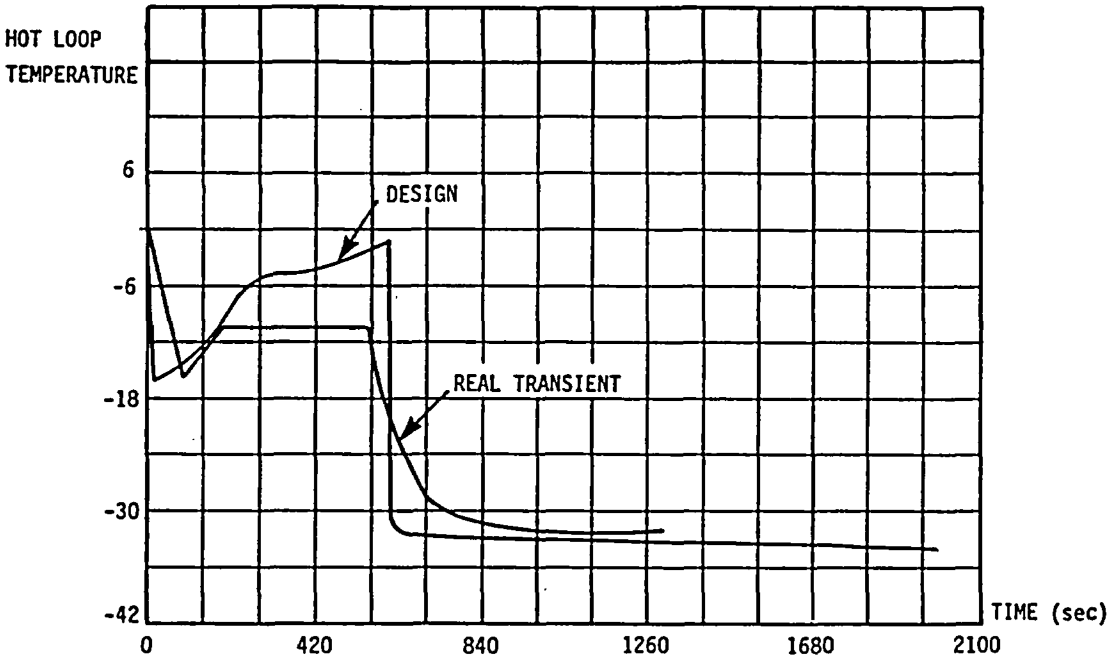


FIGURE 2 : EVOLUTION OF PRIMARY SYSTEM TEMPERATURE DURING A "LOSS OF OFF-SITE POWER" AT BLAYAIS 4 - COMPARISON WITH DESIGN HYPOTHESIS

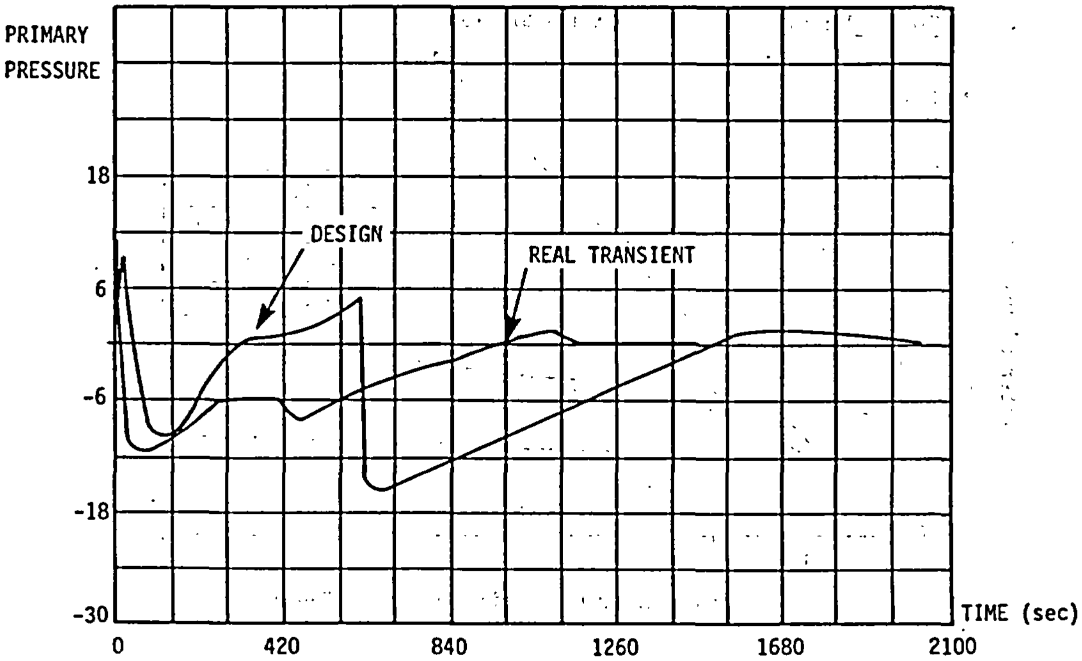


FIGURE 3 : LARGE LOAD CHANGES

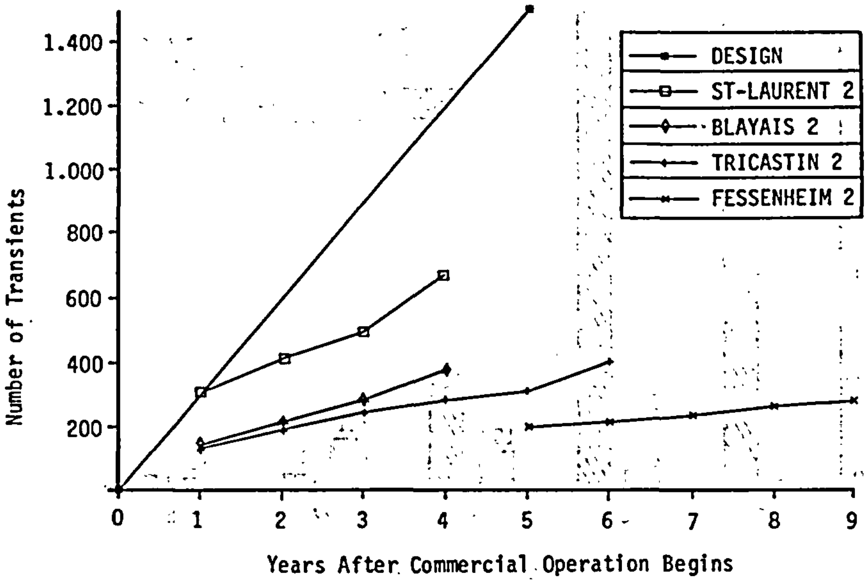


FIGURE 4 : CVCS CHARGING LINE TRANSIENTS

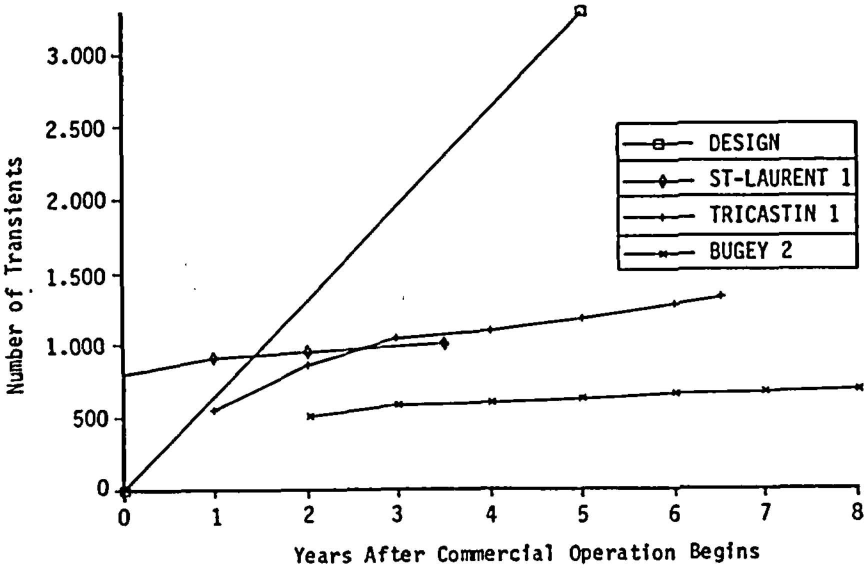


FIGURE 5 : CHARGING LINE TRANSIENTS ON BUGEY 2

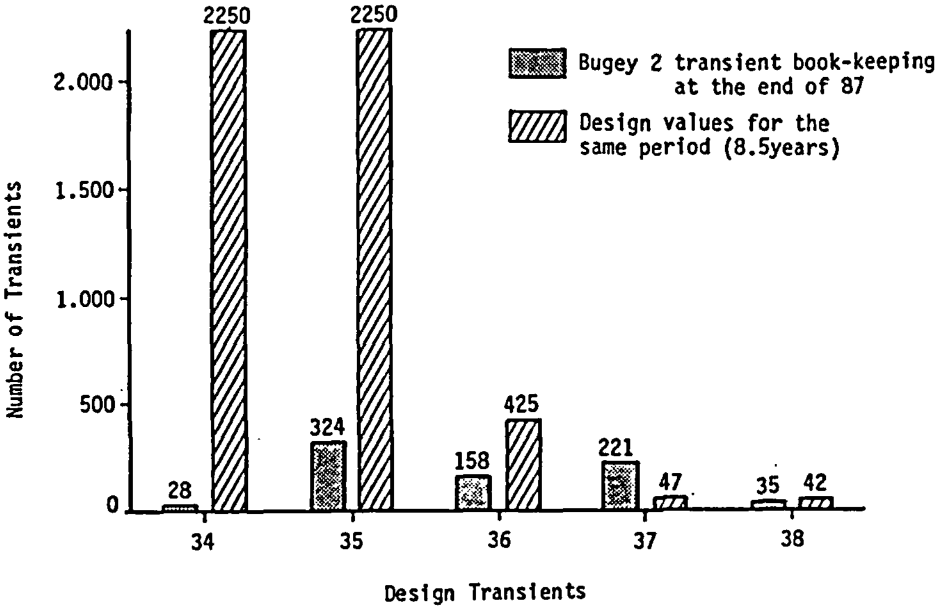


FIGURE 6 : CHARGING LINE NOOZLE USAGE FACTOR
(THERMAL SLEEVE)

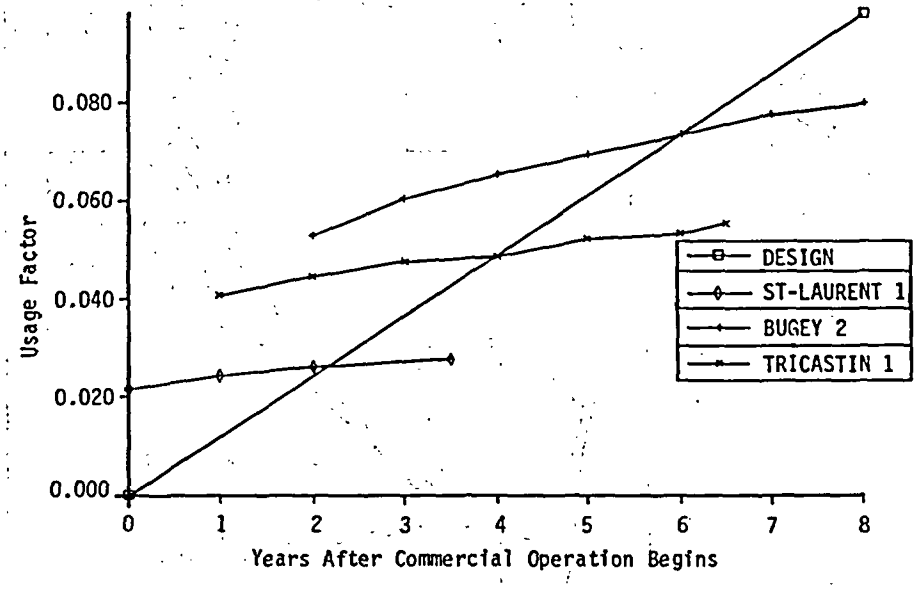


FIGURE 7 :

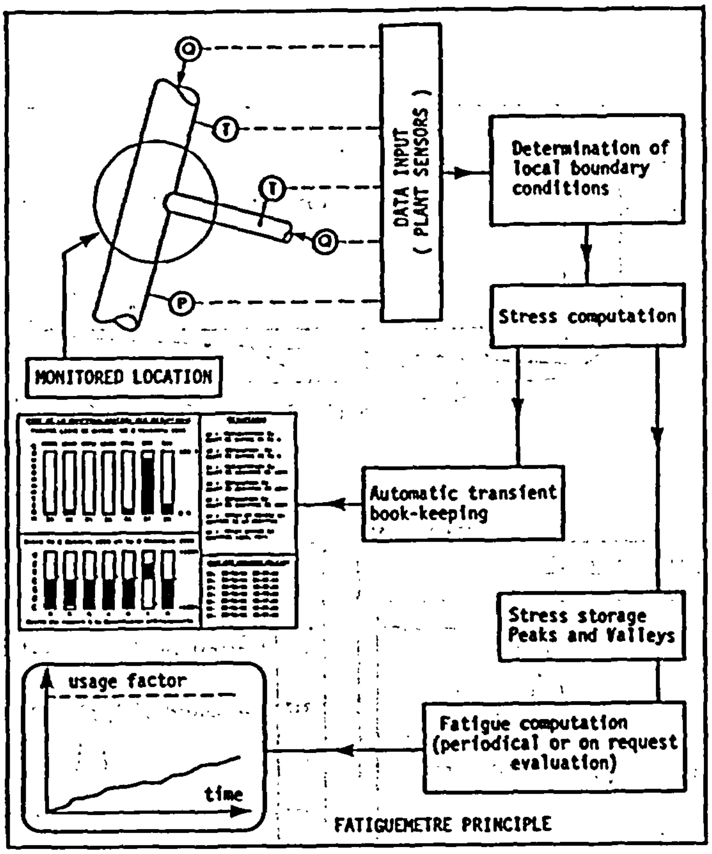
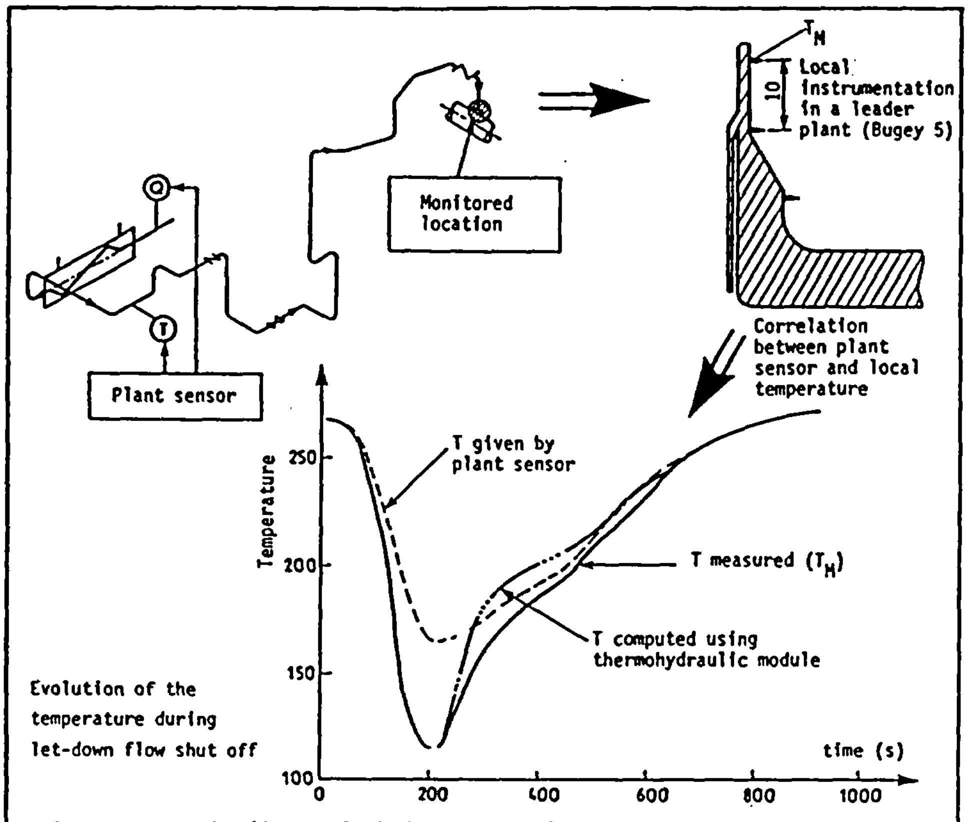
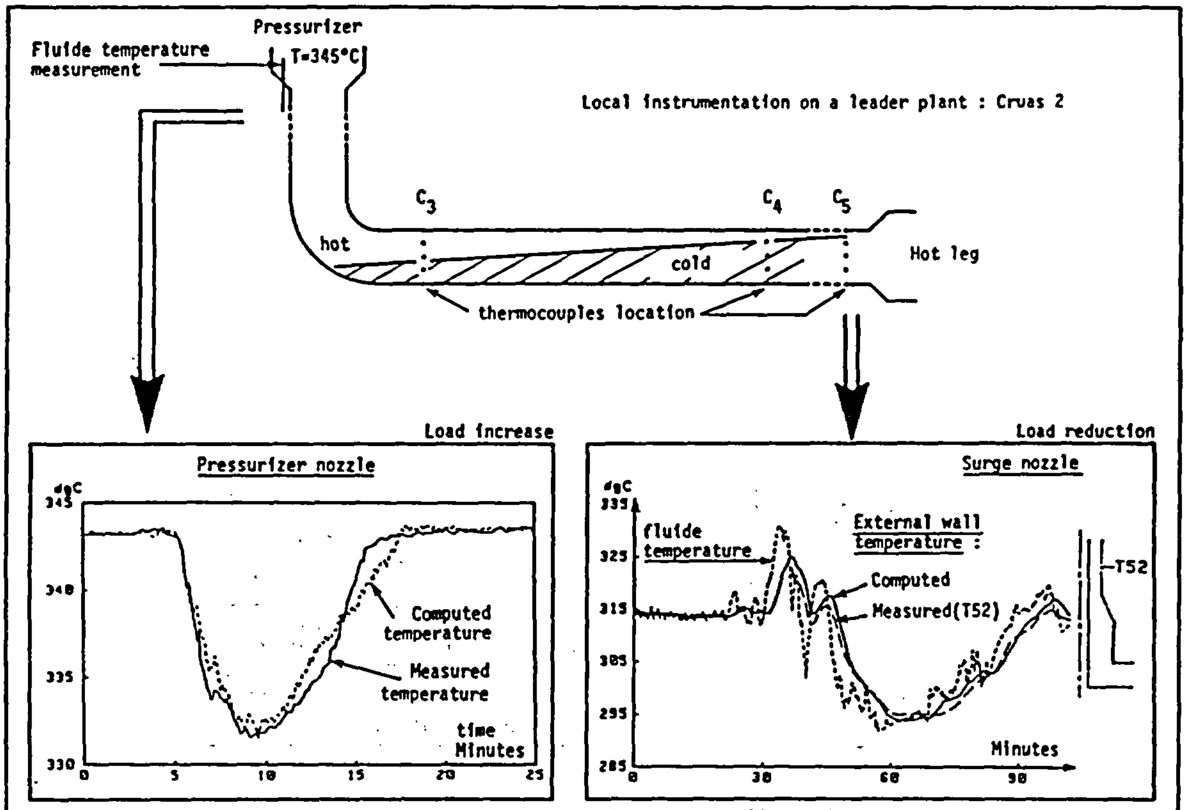


FIGURE 8 :



Bugey 2 : Charging line nozzle fatiguemeter : Validation of the thermohydraulic module

FIGURE 9 :



Dampierre 1 Fatiguemeter : Validation of thermohydraulic module

FIGURE 10 :

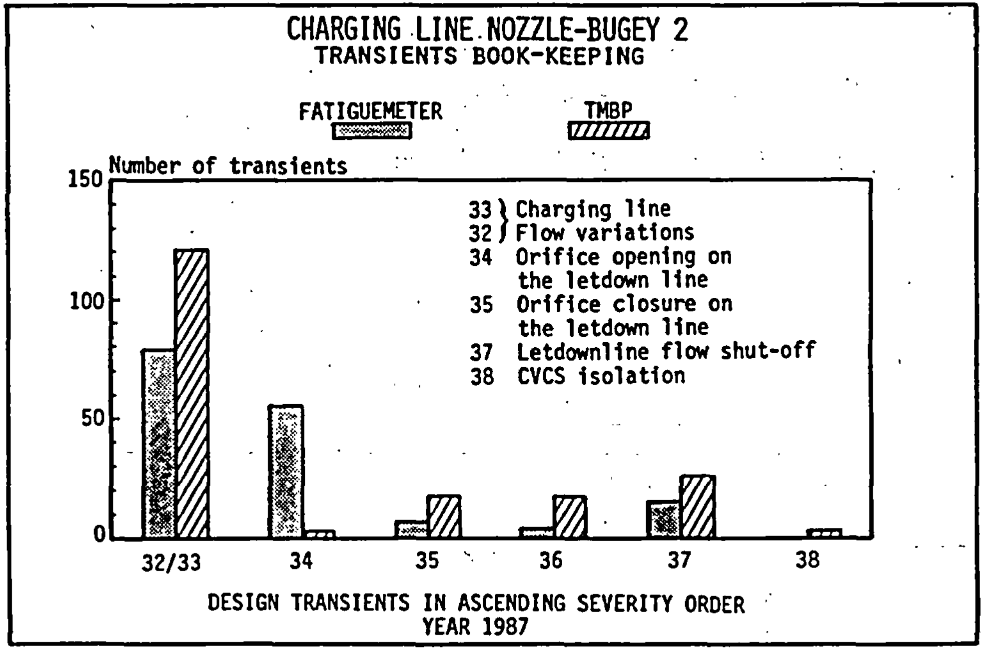


FIGURE 11 :

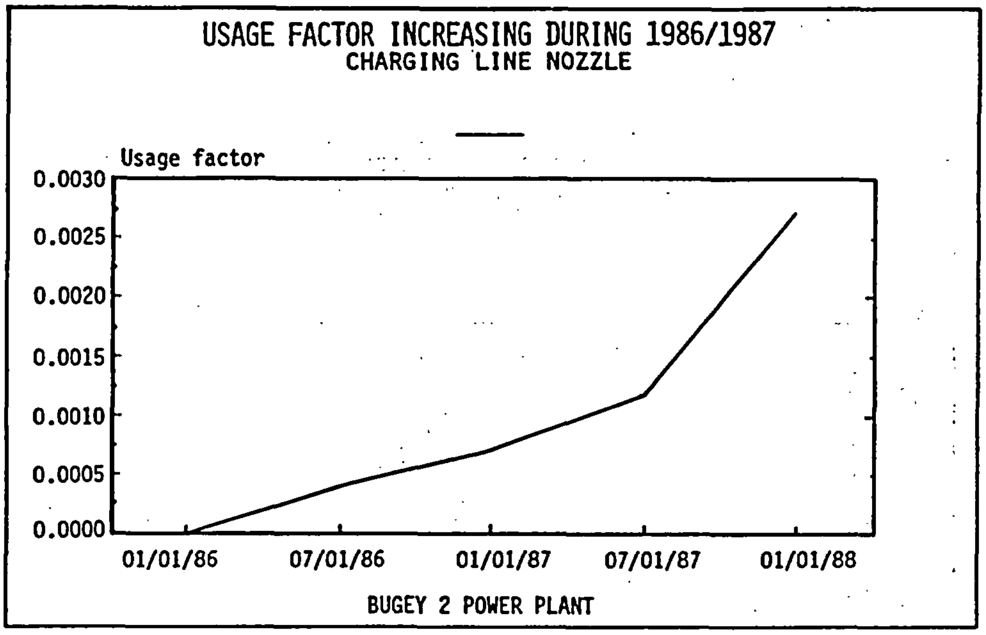


FIGURE 12 :

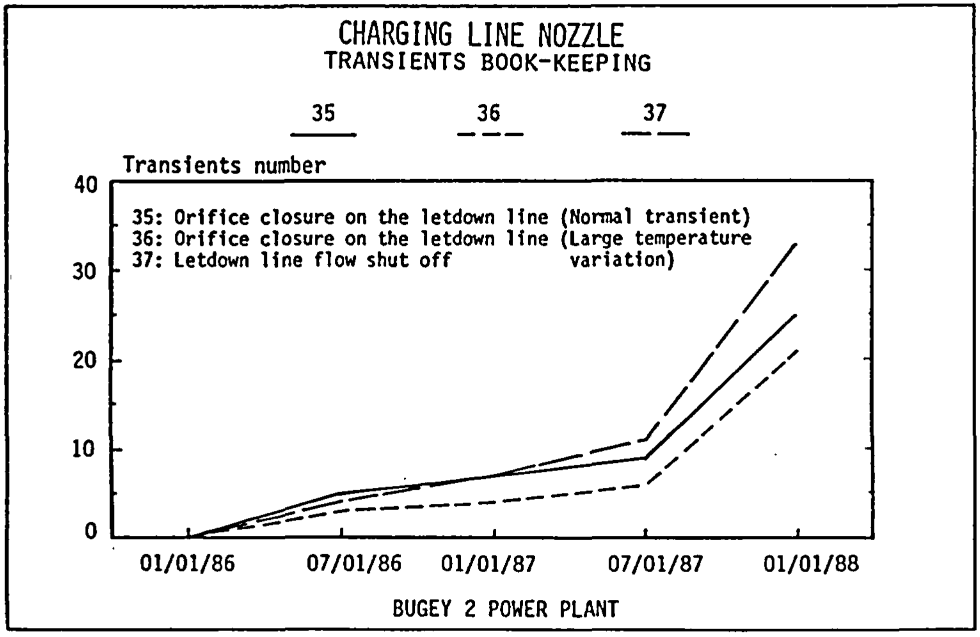


FIGURE 13 :

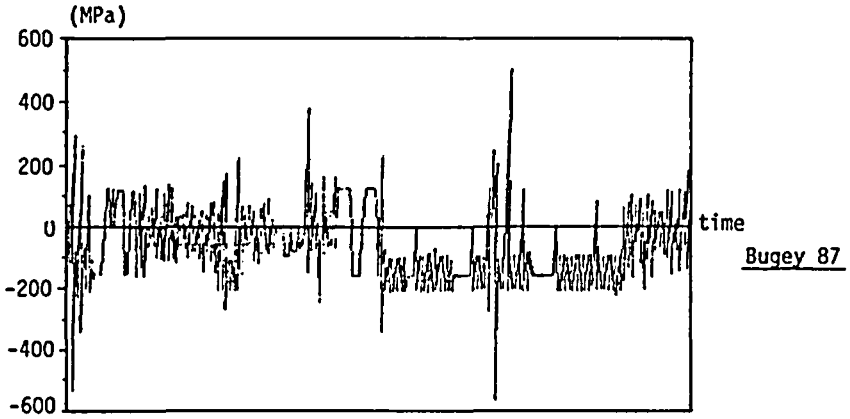
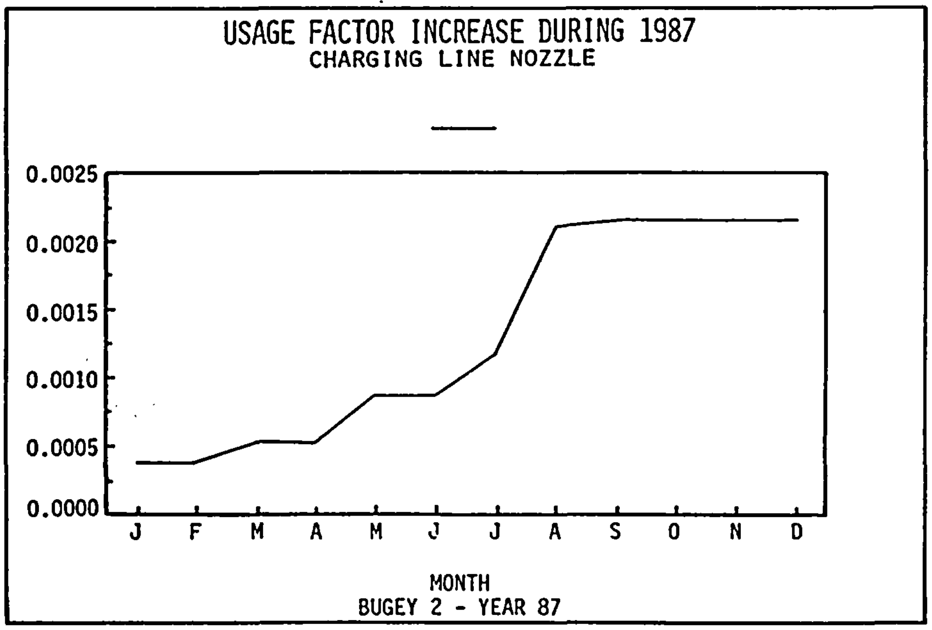
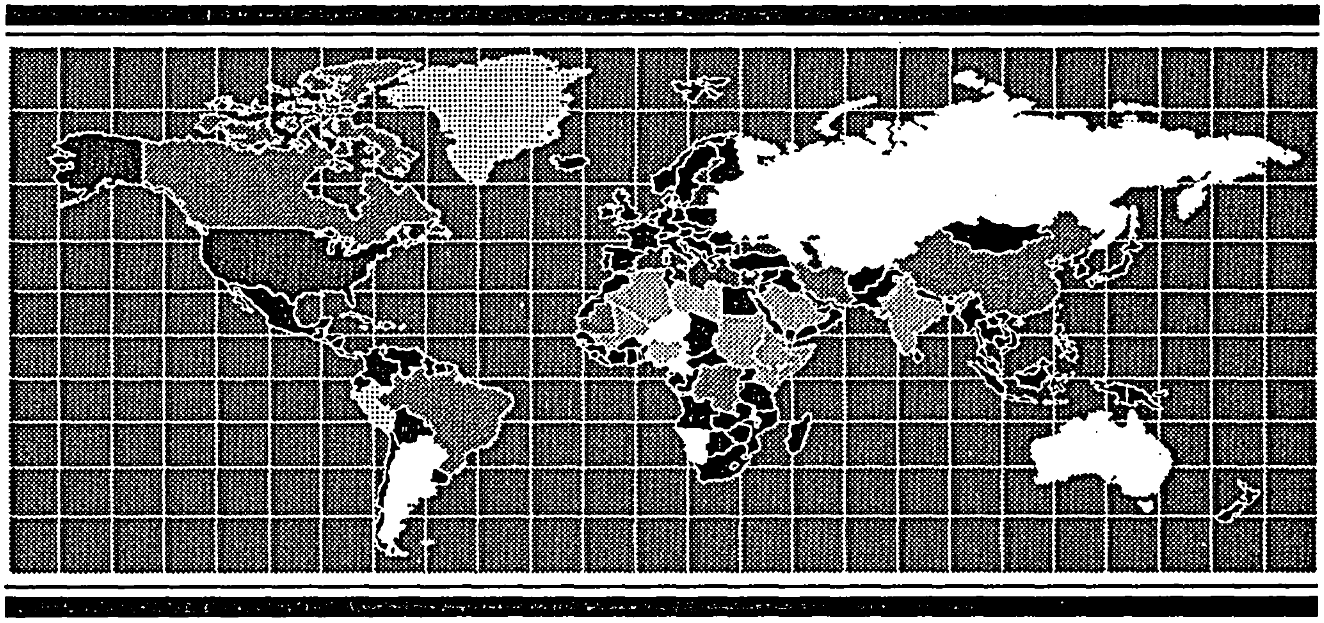


FIGURE 14 :





TECHNICAL SESSION 6

Role of Maintenance in Aging Management

August 31, 1988

Session Chairman

JACK W. ROE
Director, Division of Licensee Performance and
Quality Evaluation
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

NUCLEAR PLANT MAINTENANCE ACTIVITIES AND PLANT LIFE EXTENSION IN JAPAN

H. Mitsuda

The Kansai Electric Power Co., Inc.
Osaka Japan

1. Introduction

The nuclear power plant is very important energy source in such countries having little natural resources as Japan, and it is granted that it is one of the most reliable energy resources in view of economy, energy security and environmental protection.

Accordingly, the Kansai Electric Power Company (KEPCO) has been steadily developing nuclear power capability since the first installation of Mihama #1 unit, and we own and keep operating 9 units. A total capacity of these units has grown up to 7,408MW, occupying approximately 26% of the total power generating capacity in our power grid.

Meanwhile, the reliability of nuclear power plant has been improved considerably thanks to execution of elaborate and vigorous plant system inspection, fault detection and repair in earlier stage, improvements of inspection and maintenance system, modification of plant facilities in reference to domestic and overseas information, etc.

As a result, plant capacity factor has exceeded 70% in these several years, and plant outage rate (0.4 times per reactor year in FY 1987) has been remarkably decreased.

Under the circumstances, we have been working hard for improving plant maintenance system so as to enhance the plant reliability by means of further reduction of forced outages.

The following covers our basic concept of plant maintenance and plant life extension programs.

2. Improvement of Plant Maintenance Program

The nuclear power plant maintenance system on the part of the utilities in Japan is designed to perform the straightforward preventive maintenance of plant facilities for the purpose of operating these plants safely and reliably. In this connection, the following activities are being carrying out.

(1) Periodic inspection based upon the scheduled program for inspecting plant facilities, together with the implementation of modifications and/or repairs required

(2) Fine-meshed maintenance activities in light of the operating and surveillance records of our plant performance

(3) Special inspection based upon the operation and maintenance information obtained from other plants both at home and abroad

KEPCO has been endeavoring to improve the maintenance technology of PWR plant facilities by means of rationalizing the frequency of inspections and early detection of troubles in the evaluation of plant operation and periodic inspection data obtained from other domestic and overseas plants.

For those 5 plants being operated in our system more than 10 years, it has become necessary to look into the available predictive maintenance methods in order to maintain and enhance the reliability of plant facilities subject to aging effect.

Predicted service life of every equipment and component will be deduced on the basis of quantitative evaluation of problems experienced thus far, identifying of those problems in reference to the operating experience, design concept, and study results that had been obtained from not only domestic but also overseas plants.

Thus, such predictive maintenance system will be designed to detect any unexperienced problem in its earliest possible opportunity and determine an appropriate inspection interval as well as the opportune replacement schedule.

Consequently, we would like to establish the "predictive maintenance program" which will serve not only to maintain and enhance the reliability of the plant but also to extend the plant life.

3. Predictive Maintenance Program

(1) Current Situation

The predictive maintenance flow charts are given in Figs. 1 & 2.

As for all pieces of equipment comprising of the nuclear power plant, the safety, power output, repair cost, etc. are evaluated on priority basis as shown in Fig. 1. And, we select the important equipment.

Fig. 1 PREDICTIVE MAINTENANCE FLOW CHART

(Significant degradation on the level of components)

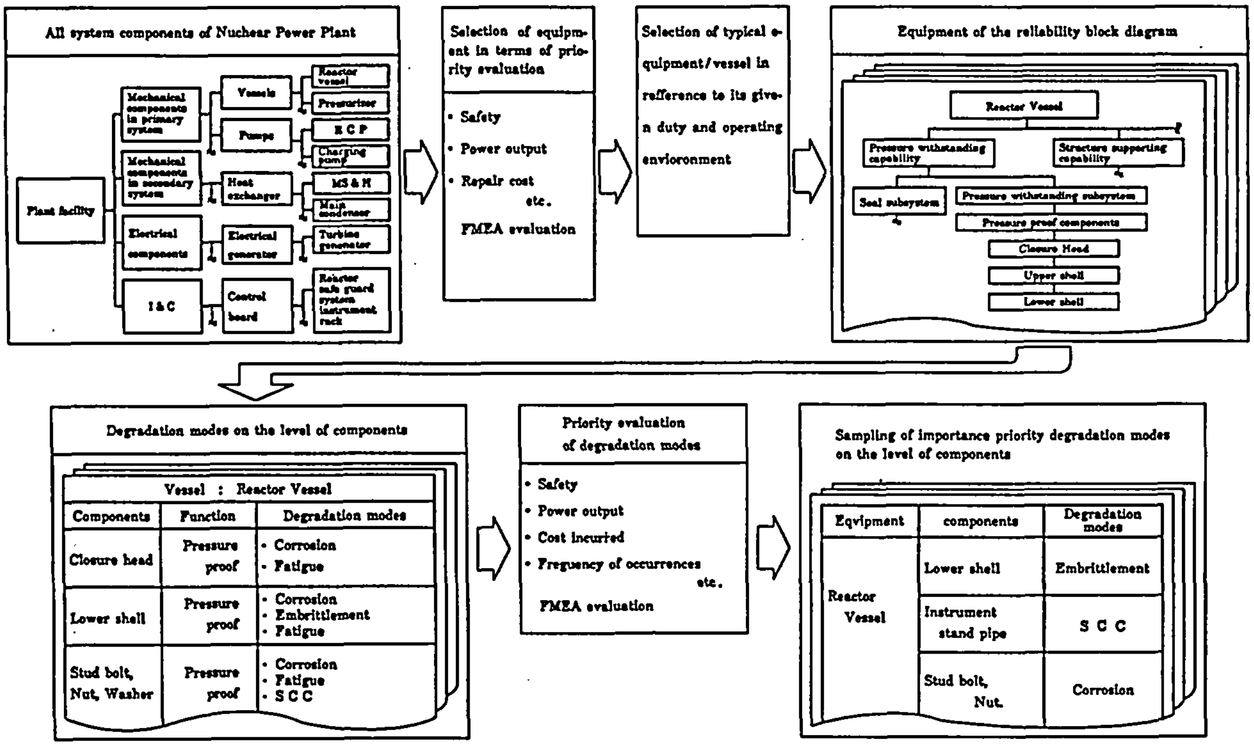
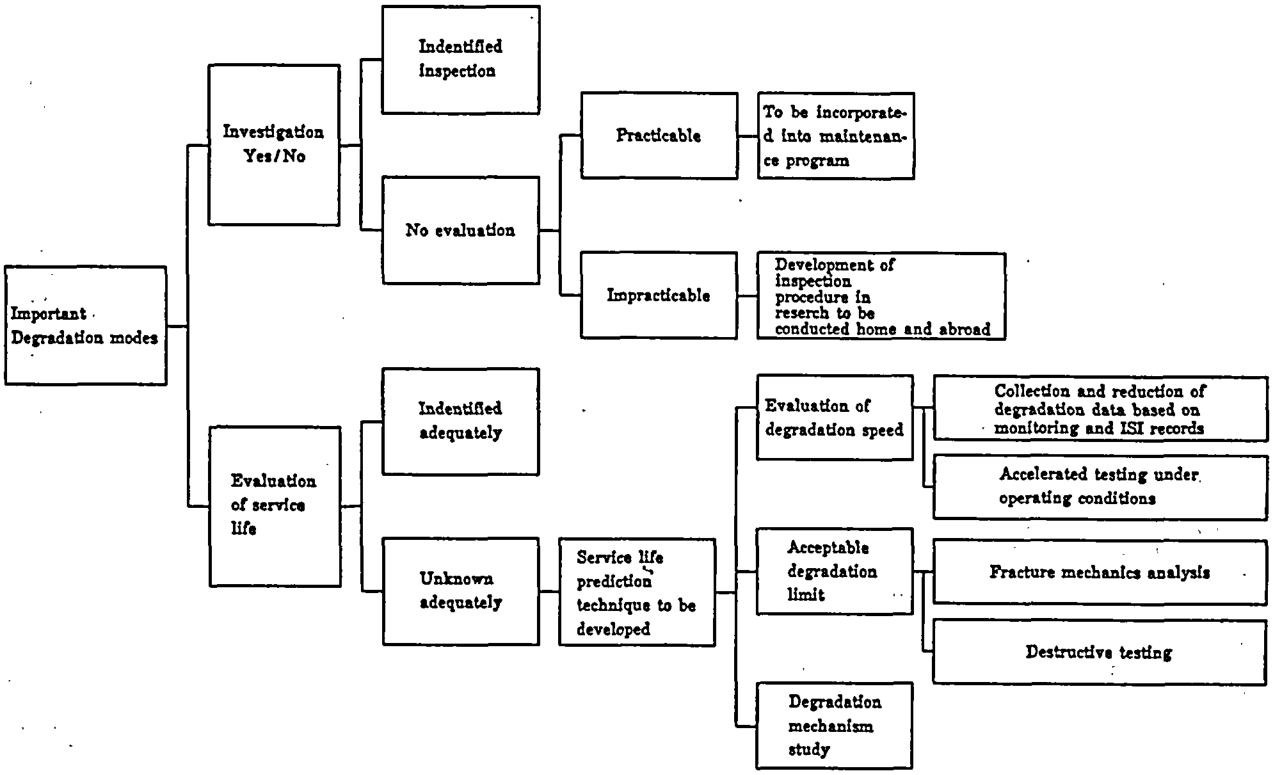


Fig. 2 PREDICTIVE MAINTENANCE FLOW CHART

(INVESTIGATION OF IMPORTANT DEGRADATION MODES)



Important equipment so selected is broken down to the level of its component by means of reliability block diagram, and degradation mode of all such components are deduced on basis of operating experiences, design concept and study results. And, we select the important degradation modes by priority evaluation.

Then, the important degradation modes will be classified as follows, together with actions against the respective degradation modes.

a) Investigation of important degradation modes.
Whenever we find current maintenance procedure capable of identifying any important degradation mode, we will try to develop the pertinent inspection procedure.

b) Evaluation of components service life.
When the propagation speed of any important degradation mode is found unknown to us, we will try to evaluate its degradation speed, acceptable degradation limit and degradation mechanism.

(2) Typical Example of Predictive Maintenance

Since 1970, such predictive maintenance program has been applied for measuring wall thinning of two-phase flow portion of the secondary piping, and, since 1985, this program has been expanded to cover single phase flow portion.

Incidentally a number of measuring points to be inspected during every annual inspection downtime is about 1,000.

On the basis of the data obtained from such measurement, the inspection frequency has been relaxed for some pipe runs.

Every excessive wall-thinned pipe run has been replaced with stainless steel pipe (See Fig. 3). Consequently, we have not as yet taken any countermeasure in particular reference to Surry #2 pipe event.

4. Plant Life Extension Program

(1) Basic Concept

The nuclear power plant is required not only to maintain its safety and reliability during the given service life but also to improve its operating economy.

It is conceivable that the service life of nuclear plant will getting longer or shorter depending upon the following two considerations; namely, the plant will last till the service life of critical pieces of plant equipment whose life extension or replacement would be extremely difficult, and even though advanced life extension or replacement schemes become available to us, the increasing maintenance cost of other system components will seriously affect the cost-competitive operation of nuclear power plant.

In case of building a new nuclear power plant, we will design it to retain its operating integrity for a period of about 40 years in due consideration of various transient and environmental conditions to be predicted from our operational experience of other plants in service.

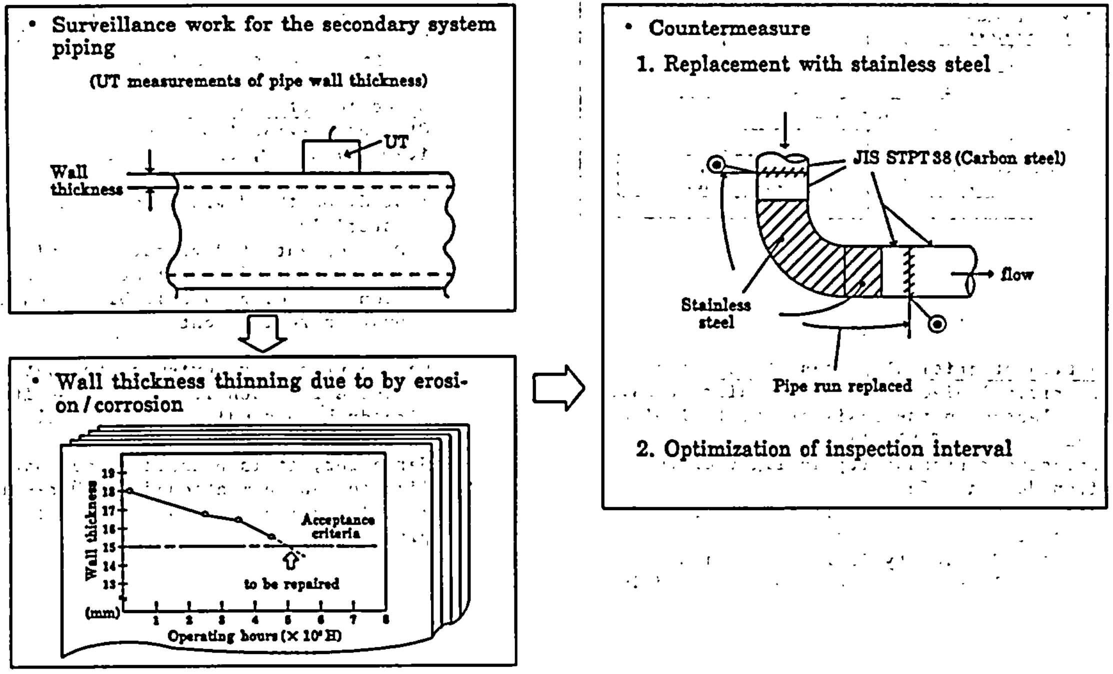
Such design practice will tell us that any new plant put on the line has sufficient operational margin in light of the findings of actual operating records and various surveillance tests. Accordingly, we believe it feasible to extend its service life beyond the given design life span, with resultant reduction in overall generation cost.

(2) Current Situation

The nuclear plant life extension program demands us to make an extensive examination and evaluation of the safety, reliability, and economy of the plant, and a considerable long term will be required for this activity. In other words, the program must be executed systematically and effectively.

Fig. 3 TYPICAL EXAMPLE OF PREDICTIVE MAINTENANCE

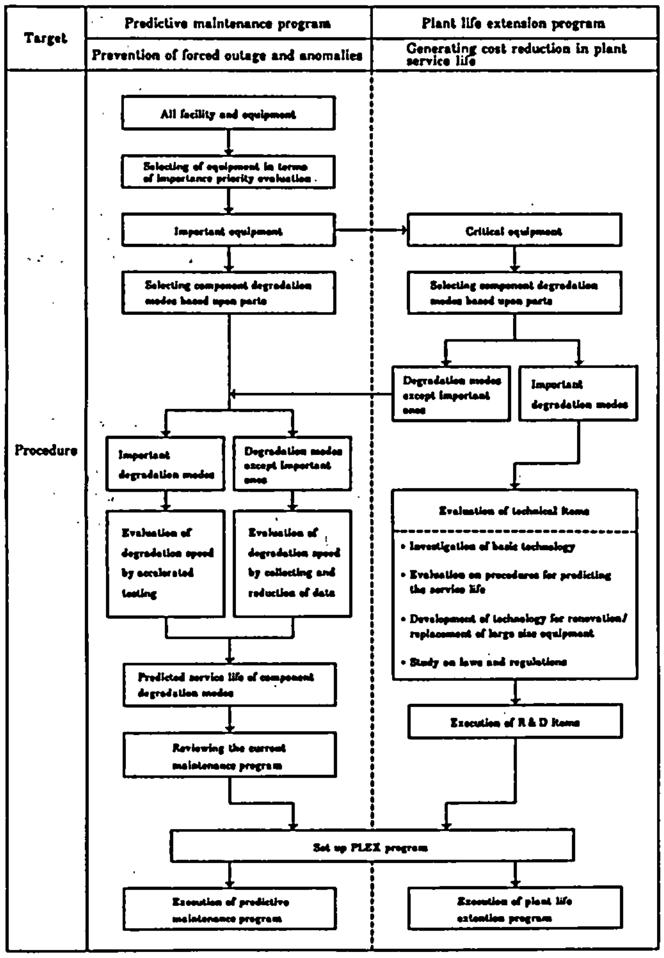
(the secondary system pipe wall thinning problem)



We have been systematically studying plant life extension technology for about five years since 1985, with the objective of providing an insight into the feasibility of nuclear plant life extension. Incidentally, we are given to understand that the studies on nuclear plant life extension have been vigorously carried out in U.S.A. and Europe.

In view of such global trends, it is required for us to expedite our study on the plant life extension technology in accordance with the sequential procedures as shown in the right-hand column of Fig. 4.

Fig. 4 RELATIONSHIP BETWEEN THE PREDICTIVE MAINTENANCE VS. PLEX PROGRAM



a) Selection of critical pieces of equipment.

Most critical pieces of plant equipment have been selected on the basis of the following evaluation (See Table 1):

- Replacement probability in light of the given design life
- Scope of renovation/replacement work and associated exposure dose
- Relative difficulty in field application of technology available at present and in future

Table 1 Examples of critical equipment

No.	Equipment
1	Reactor vessel
2	Pressurizer
3	Steam generator
4	Primary coolant pump
5	Reactor vessel internals
6	Main coolant pipe
7	Control rod drive mechanism
8	Containment vessel
9	Cables (inside containment vessel)

b) Identification of degradation modes.

First, every critical equipment will be broken down to the level of components. Then, every component life will be estimated based upon the evaluation of all probable degradation modes to be deduced from empirical knowledge and data as well as study results.
And, all important degradation modes which are deemed influential to equipment life will be chosen.

[Examples of important degradation modes]
Fatigue, Embrittlement, Corrosion, SCC, wear, etc.

c) Technical evaluation of degradation modes.

All important degradation modes so selected will be sorted out for further study in due consideration of the examination results of pertinent technical papers covering the following:

- Basic study on structural material degradation and so on
- Studies on equipment life evaluation
- Studies on life extension technologies applicable to renovation/replacement

d) Selection of R & D items.

Of those degradation modes as referred to in item (c) above, the following are selected for R & D programs:

- Study on the degradation of equipment and materials
- Development of degradation monitoring system
- Development of equipment service life extension technology applicable to equipment renovation/replacement

We are presently reviewing the detailed study schedule for each item, and some items have already been studied. The examples are shown in Table 2.
Hereafter, it is planned on our part to utilize the above-mentioned R & D results for roughly estimating the service life of nuclear plant.

Table 2 Examples of R & D Items

Material Test	• Material test of low alloy steel under the primary water environment • Research on low alloy steel embrittlement • Research of stainless steel thermal aging • Research of stainless steel embrittlement
Monitoring	• Development of material fatigue monitoring syste • Development of material degradation monitoring system • Development of AE monitoring system • Development of rotating equipment monitoring system
Life extension technology	• Development of renovation/replacement technology for large size equipment • Development of degradation alleviating measure

5. Predictive Maintenance vs. Plant Life Extension Program

We at KEPCO have been working out the predictive maintenance program for all of the plant equipment, in which every critical equipment are included in the scope of plant life extension program. (See Fig. 4)

Therefore, this approach will eventually allow us to sort out all the study items indispensable for successfully carrying out the predictive maintenance program. In view of the fact that our government is presently supporting domestic and international projects dealing with certain domains of plant life extension technology, we are planning to actively participate in such activities being carried out at home and abroad for the purpose of enabling us to fully enjoy the fruits of such studies.

6. Future Tasks

In order for us to promote the foregoing R & D activities, we must actively tackle the following tasks:

a) Establishment of Conceptional Guideline for Pursuing Plant Life Extension Program

The study scope and elements of the plant life extension program will vary greatly depending upon how to appreciate the significance of "safety" and "economy", which will tell us how it is important for us, all the parties concerned to establish such conceptional guideline.

b) Drafting of a Master Plan

The foregoing activities have been getting intensified on the part of many industrial organizations, and R & D items are diversified. Therefore, we believe it necessary to set up a systematic program so that an effective study may be carried out by every party.

c) Establishment of Equipment Service Life Prediction Technique

In order for us to extend the plant life, it becomes indispensable to predict the equipment service life with most reliable prediction technique to be developed.

d) Model for Predicting the Plant Life

In addition to the life prediction technique mentioned in item (c) above, we need a model for predicting the plant life in due consideration of the safety, reliability and economy of total plant.

e) International Cooperation

The United States is ahead of Japan in the field of R & D program mentioned above, and the same programs are likely to be actively pursued in other countries.

In light of global nuclear industry interest, it would be mutually beneficial for the parties concerned to closely communicate and exchange the pertinent information in the field of predictive maintenance technology evolution.

7. Postscript

In the foregoing, we describe the process for reviewing the advanced technology for nuclear power plant predictive maintenance program, together with its current situation.

As it will take a long time to establish such technology, we are to efficiently conduct such program in a phased approach.

It is believed on our part that the predictive maintenance technology established will effectively contribute to the enhancement of in-service plant reliability and the extension of plant life, and also to the safety, reliability and economy of every new nuclear power plant.

SIMULATION OF MAINTENANCE EFFECTIVENESS IN REPAIRABLE SYSTEMS TO CONTROL RISK DUE TO AGING[a]

D. G. Satterwhite, N. G. Cathey, B. M. Meale, P. E. MacDonald
Idaho National Engineering Laboratory

ABSTRACT

Time- or cyclic-dependent degradation (aging) processes such as wear, corrosion, and fatigue have caused problems at some of the U.S. reactors. This paper presents a methodology to develop optimum maintenance programs that would help control aging-related risks. The study addresses the predictive behavior of systems containing repairable components as represented in current probabilistic risk assessment (PRA) models. The system model chosen for the study is the Draft NUREG-1150 Surry auxiliary feedwater (AFW) system. The simulation results provide an indication of the periodicity at which the aging effects must be corrected to maintain the system failure probability at or below the calculated levels. When maintenance is allowed to periodically correct the effects of aging, the system failure probability is essentially constant over 60 years. Also, results indicate that the system unreliability is not linearly related to the periodicity of aging correction. Therefore, a plant maintenance program would have significant latitude in developing the desired maintenance periodicity limits. In fact, the calculated values for component maintenance periodicity are of such time intervals that aging control can easily be accommodated by a plant maintenance program.

1. INTRODUCTION

Time- or cyclic-dependent degradation (aging) processes such as wear, corrosion, and fatigue have caused problems at some of the U.S. reactors. These problems have raised questions about the degradation of safety equipment at the plants and the impact of such degradation on the public. However, many of the aging issues have been and are being addressed by the nuclear industry through research, improved designs, standards development, and, especially, improved operation and maintenance practices. Nevertheless, components will continue to degrade, and it is important to establish a method to quantify and control the risks associated with this degradation. Maintenance practices have the greatest impact on controlling aging. This paper presents a methodology that can be used to develop optimum maintenance programs that would help to control aging-related risks.

This effort is part of the U.S. Nuclear Regulatory Commission (NRC) Nuclear Plant Aging Research (NPAR) Program that is being conducted at the Idaho National Engineering Laboratory (INEL) and other DOE laboratories. The methodology that has been developed, coupled with other efforts to quantify the failure characteristics of aging components, will provide a means to quantify time-dependent risks and a basis for establishing optimized maintenance programs for controlling these risks. The use of this method depends on establishing safety goals for either total plant risk (core melt frequency) or system reliability. These goals can be established through the use of a plant probabilistic risk assessment (PRA). Then, maintenance programs can be formulated to assure that the potential unavailability of individual systems remain in accordance with the desired reliability goals.

A simulation analysis is used to quantify the time-dependent unreliability of the AFW system and demonstrate how a maintenance program can be developed that will control the aging risk to levels below an established goal. The analysis includes calculations of the time-dependent system failure probability as a function of postulated testing and maintenance (T&M) practices. This provides a way in which a periodicity of necessary corrective activities can be properly established. Finally, this paper discusses how the methodology can be used to develop instruments to assess the effectiveness of T&M programs in addressing the aging effects present in the operational environment of the components.

The methodology presented in this paper can be readily extended to assess aging degradation and T&M effects on core melt frequency. However, in order to take advantage of existing aging models and data, the scope of this study has been restricted to a particular system. The system model chosen for the simulation analysis is the Draft NUREG-1150 Surry auxiliary feedwater (AFW) system.[1] Using the Surry AFW system, this study addresses the predictive behavior of systems containing repairable components as represented in current PRA models. This study does not consider structural components since techniques have not been developed to model structural aging degradation. However, once the modeling capability has been developed, the techniques discussed in this paper can be applied to structures.

2. PRA MODEL

The PRA fault tree model used in the current analysis is based on the Draft NUREG-1150 Surry AFW system model obtained from Reference 1. The level of detail of the AFW model has been extended so that all the constituent component faults in the pipe segments were modeled. The modeling extension was performed as part of the work scope for the NRC Plant Risk Status Information Management System (PRISIM) program. A system schematic with component identifiers is presented in Figure 1. Component fault descriptions and aging data are presented in Table 1.

Draft NUREG-1150 failure rate data for the Surry AFW system was used as the basis for calculating the time-dependent failure rates in this analysis. The Surry failure rates are generally expressed as per demand failure probabilities with very few hourly

---

[a] Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570.

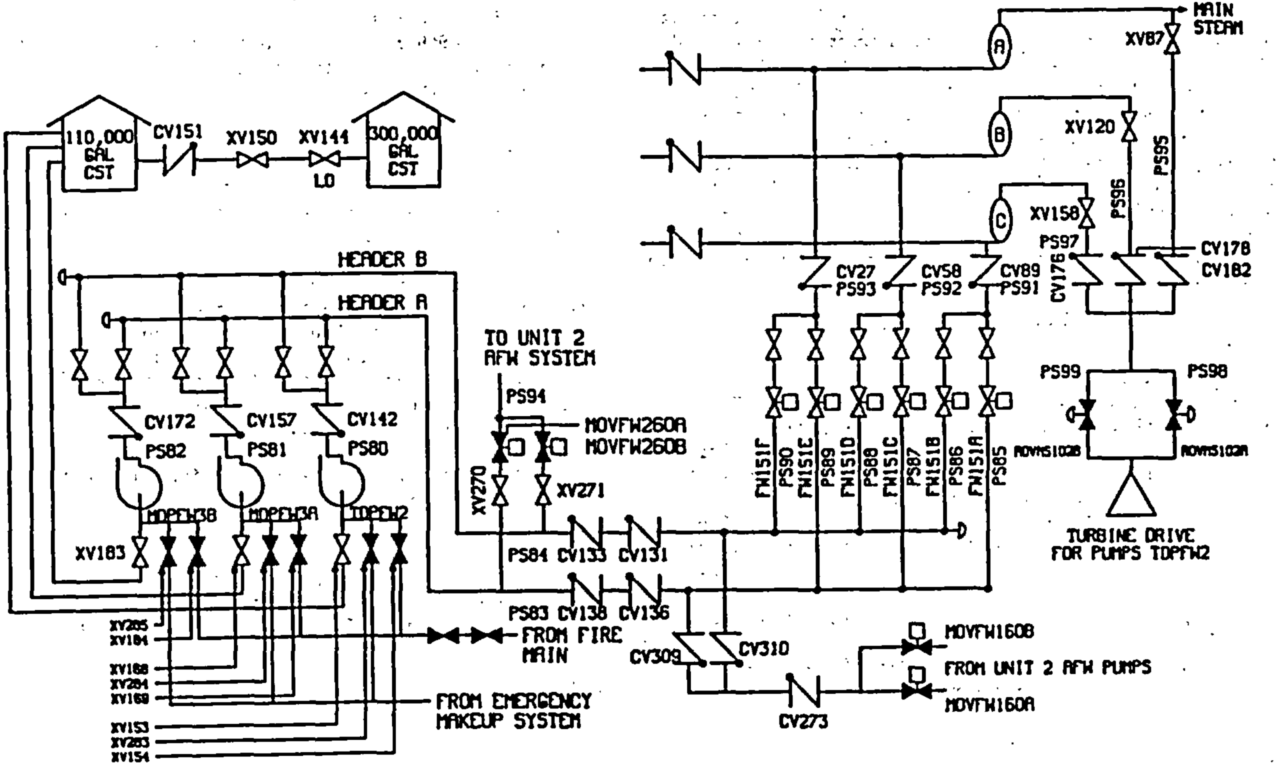


Figure 1. Auxiliary feedwater system schematic.

TABLE 1. COMPONENT FAULT FAILURE AND AGING RATE DATA

Component Fault Type	Testing Internal (hr)	Median Failure Rate Per Hour	Error Factor	Mean Failure Rate Per Hour	Aging Fraction	Aging Rate Per Hour
Air-operated valve Loss of flow (AOV)	730	2.2E-5	3	2.75E-5	0.68	2.35E-9
Check valve back-leakage Casing AFWS pump steam binding (STMBD)	8	--	--	1.8E-5	0.91	1.83E-9
Check valve Failure to open (CKV)	730	2.74E-7	3	3.42E-7	0.90	1.47E-10
Motor-driven pump Fails to run (MDP-FR)	730	5.5E-8	10	1.46E-7	0.45	6.7E-12
Motor-driven pump Fails to start (MDP-FS)	730	1.5E-5	2.2	1.68E-5	0.44	3.0E-10
Turbine-driven pump Fails to run (TDP-FR)	730	1.4E-7	10	3.73E-7	0.69	2.3E-11
Turbine-driven pump Fails to start (TDP-FS)	730	2.0E-5	4.6	3.08E-5	0.62	2.1E-9

failure rates present in the data. Since the analysis is concerned with the time-dependent effects of aging and its control, all failure rates must be expressed as hourly rates. The demand failure probabilities were converted to hourly rates by estimating the number of demands expected during a testing interval and then normalized by the number of hours in the interval. The Draft NUREG-1150 failure rate data expressed in hourly rates such as "pump failure to run" was multiplied by the expected mission time during a testing interval, then normalized by the interval. The base data and results of the conversion to hourly rates are presented in Table 1.

3. AGING FAILURE RATE MODEL

Aging causes components to degrade at various rates and exhibit various failure behaviors. The component failure behavior can be modeled as a function of time. The time-dependent component failure rate can be linear or nonlinear with age, represents a continuous aging process or a cyclic-dependent process, or be related to a threshold effect. Ideally, the individual stressors should be explicitly modeled with failure rate models for each unique stressor. However, due to the sparseness of data for use in developing explicit failure rate models, it is not currently possible to statistically distinguish the individual stressor behaviors. Therefore, the time-dependent failure rate model used in this analysis represents an aggregate of the aging-related failure characteristics for individual stressors.

The time-dependent failure rate model chosen for use in the current study is the linear aging model developed by Dr. W. E. Vesely in Reference 2. This failure rate model was developed for use in deriving time-dependent failure rates from the existing operational failure data bases compiled by the NRC and the nuclear industry. The simplicity of the model is appropriate since the sparseness of the data will not necessarily support more sophisticated modeling of component failure rates. Other failure rate models are under development but were not available for this study. Due to the exploratory nature of this study, the linear aging model should be adequate to demonstrate the methodology for developing a maintenance strategy. More sophisticated aging models can be incorporated as they become available.

The time-dependent failure rate derived in Reference 2 has the following form

$$r(t) = R_f + \alpha t \tag{1}$$

where

R_f = the portion of the failure rate resulting from random failures

αt = the linear time-dependent portion of the component failure rate where α is the aging acceleration rate, termed aging rate, resulting from component degradation by aging.

The event data used to calculate failure rates used in PRAs contains both time-dependent and random failures. In order to model component aging, the random portion must be extracted from the traditional PRA failure rate. The random portion is extracted by determining the portion of failures that are due to nonaging causes recorded for a component in the event data bases. This is done by examining failure event reports for individual component types within a system to assess the number of aging-related versus total component failures (aging-related is used because of its importance in other applications). This evaluation yields an aging fraction, f_a, which is then used to calculate R_f by

$$R_f = R_T\ (1 - f_a) \tag{2}$$

where R_T is the total component failure rate as obtained from a PRA.

The aging rate can be estimated by several different methods. The method chosen for this study is the Cox partial likelihood estimation used in hazards analysis detailed in Reference 3 and extended in Reference 4. This method was judged to be more accurate than an estimation method such as the moments method presented in Reference 2.

The Cox partial likelihood function employing the linear aging model is expressed as

$$L(\alpha) = \prod_{i=1}^{n} \frac{\alpha T_i}{\alpha T_i + \lambda_o} \cdot \prod_{j=1}^{m} \frac{\lambda_o}{\alpha T_j + \lambda_o} \tag{3}$$

where

α = the aging acceleration rate

T_i = i^{th} aging related failure time resulting from one aging mechanism

T_j = j^{th} random failure time

λ_o = random failure rate.

The current study uses Equation (3) to calculate the aging rate for specific failure modes of the components modeled with time-dependent failure characteristics. While the equation is written for a single aging mechanism, the aging rate can be generalized to accommodate multiple mechanisms, if the mechanisms are acting on approximately the same time scales. Equation (3) is solved for the maximum likelihood value of the aging rate, α, by first taking the Ln_e of both sides, differentiation with respect to α, and solving for the maxima using recursive techniques.

4. FAILURE RATE DATA

The data used to calculate the component aging rates were extracted from the Nuclear Plant Reliability Data System (NPRDS). The AFW data was the result of the aging failure-cause identification study reported in Reference 5. NPRDS failure records were examined in this study to determine the reported cause of failure, failure mode, age of the component, and aging classification at the time of failure. The aging classification (using a structured approach detailed in Reference 5) is divided into three categories: aging, nonaging, and unknown. The unknown category is used for failure records that do not contain enough information to provide an accurate aging classification. No attempt was made to remove the effects of actual testing and maintenance activities conducted on the components whose failure histories provided the data used to calculate the aging rates.

There are several sources of error propagated through the data acquisition and aging rate calculation process. These are (a) error in the reporting (not quantifiable) that can result from the misinterpretation of the cause of failure or erroneous component age, (b) error in analyzing the data (not quantifiable), which, for example, can occur if a random maintenance error causes very accelerated wearout of a component; in which case, the failure would be classified as aging unless it were known that the random error caused the failure, and (c) error in fitting a linear model to the data through a likelihood estimation process (can be estimated). To aid in minimizing these sources of error, the failure data classified as "aging unknown" are used to provide an aging upper- and lower-bound estimation of the aging rate. This study used the upper bound of the aging rate for conservatism. Results of the aging rate calculations are presented in Table 1.

The data presented in Table 1 reflect all events for which it was possible to model aging effects. Many basic events present in the AFW model were modeled as nonaging (human related, T&M events, and actuation failures). The information necessary to model aging characteristics for these types of events does not exist at this time. Additionally, Reference 5 contained no failure data for plugging of valves. Therefore, the basic events in the AFW related to valve plugging could not be modeled with aging characteristics. Electrical system failures are also represented in the AFW model but were not allowed to age since only the aging characteristics of the AFW components are being evaluated. Of necessity, more detailed studies at the core melt frequency level would contain the aging modeling of all safety and support systems.

5. AGING EVALUATION SIMULATION MODEL

The simulation process used to calculate the time-dependent AWF system failure probability is depicted in Figure 2. This process utilizes cut set equations, time-dependent component failure probabilities, and Monte Carlo component failure modeling to assess system failure probability response to maintenance activities.

The aging simulation model can use any applicable time-dependent component failure rate model. For this study, the linear aging model has been used to calculate component failure probability since it has undergone the most development to date. The simulation model was developed to calculate time-dependent system unreliability for use in studying T&M effects. The plant system aging and reliability characteristics can be determined by simulating different T&M options coupled with the failure rate modeling. The first T&M option of interest is repair, in which each T&M activity returns the component to a "good as old" condition. For this option, the aging rate will not be controlled or reset by any T&M activity. Therefore, the time-dependent portion of the failure rate continues to increase with component age. This option is judged to be a very conservative representation of actual maintenance activities. The second T&M option of interest would allow component failure rates to be modified on the basis of a rule based system influenced by T&M activities. In this study, the simulation model uses component failure (determined by Monte Carlo techniques) as the trigger to modify that particular component's failure rate. For the purposes of demonstration, the only modification allowed to the component failure rate was that the time-dependent portion of the failure rate could be reset to zero, thus simulating refurbishment or "good as new" maintenance. The aging rate remains unchanged, but the time scale is reset to represent refurbishment or replacement. In general, a rule based system can be developed to account for various levels of maintenance activity. However, the development of such a rule based system is beyond the scope of this study.

The time-dependent component failure probability can be expressed as

$$F(t) = 1 - \exp\{-\int r(t)dt\} \tag{4}$$

Substituting Equation (1) for r(t), and integrating using appropriate integration limits, it can be shown that the instantaneous component failure probability is expressed as

$$F(t) = 1 - \exp\{-R_f(t-T_{1w}\} \cdot \exp\{-(\alpha/2)[(t-T_{1r})^2 - (T_{1w}-T_{1r})^2]\} \tag{5}$$

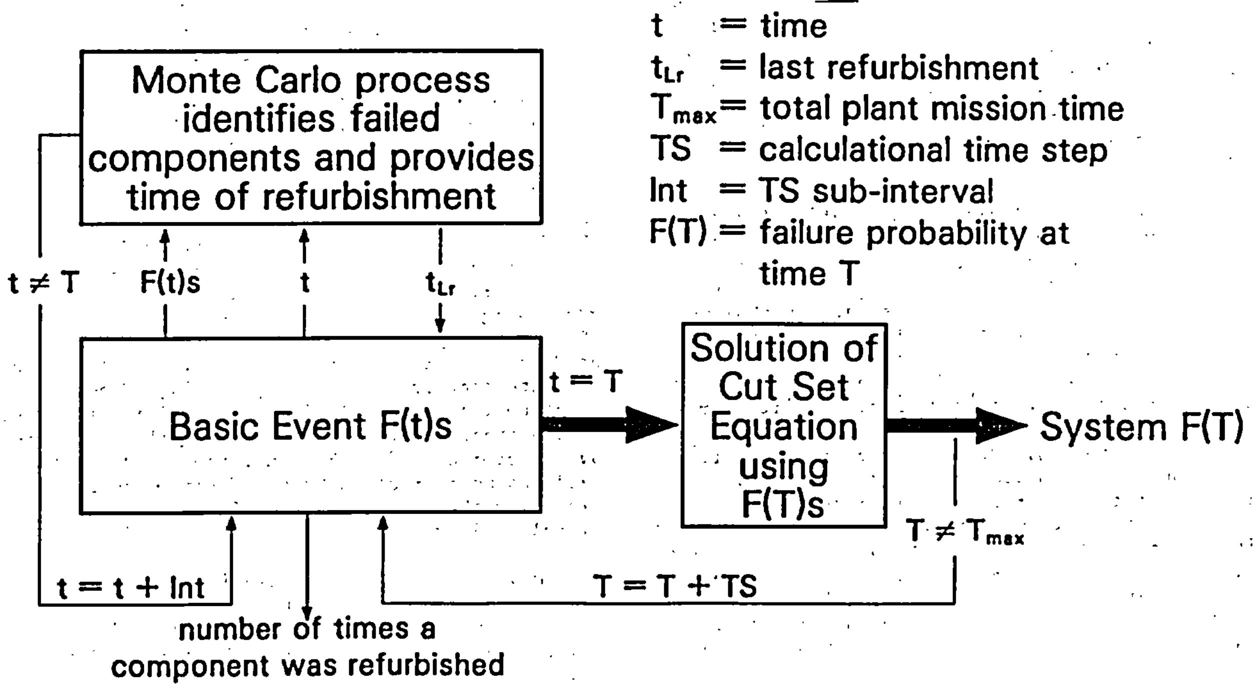


Figure 2. Simulation process

where

T_{lw} = the last known time that the component was working

T_{lr} = the last time at which the component was refurbished

The failure probability calculated using Equation (5) can be considered to express component unavailability. However, it is more properly termed unreliability since it lacks the nonzero contribution of repair time for failed components. This is judged to have little impact due to the comparative nature of this study but would be included in more detailed studies. In this study, the times of interest occur at the point of maximum failure probability, which occurs at the end of testing intervals. Therefore, $t-T_{lw}$ becomes T_{int}, which corresponds to the assumed testing interval for components in the AFW system. Except for certain check valves that are monitored for back leakage each shift, this interval is assumed to be one month.

Ideally, as a component ages and is maintained by T&M activities, the aging rate is expected to change. Based on the details of a maintenance action, a rule based system for calculating maintenance effectiveness would allow modification of the aging rate as a plant ages. This rule based system would be of interest in monitoring the failure probability of the system as the plant ages. However, since this study is predictive and a rule based procedure can not be used unless detailed maintenance activities are considered, the time-dependent effects are incorporated by adjusting the time scales used to represent the component's service age.

Equation (5) is used in this study to simulate component repair by setting $T_{lr} = 0.0$. In this simulation mode, any test or maintenance is assumed to return the component to operation with no improvement in the component degradations resulting from aging. Therefore, the aging scale will not be modified. This simulation mode is depicted by Figure 2 when the Monte Carlo process is inactive. Since maintenance addressing failure results in repair and aging scales are not reset, there is no need to identify specific component failures. The simulation process proceeds by calculating component failure probabilities and using the AFW cut set equation to produce system failure probability at each time of interest (calculation at time step, TS) until the maximum plant mission time, T_{max}, is reached.

The simulation model also uses Equation (5) to model effective maintenance by assuming refurbishment instead of repair for failed components. In this simulation mode, see Figure 2, the T_{lr} term of Equation (5) is determined using a Monte Carlo technique, modeling component failure to reset the aging scales when estimating system failure probability. The Monte Carlo technique used by the simulation consists of comparing the calculated failure probability for each component at a given time, t, to a random number ranging from 0.0 to 1.0. During the comparison, if the random number is smaller than the calculated failure rate at time t, the term T_{lr} is set to t, thereby simulating a refurbishment or replacement activity and correcting the effects of the aging. The simulation continues this process for each component failure event at subintervals corresponding to testing and maintenance intervals until a calculational timestep is reached. At this point, the system failure probability is produced from the cut-set equation. The process is continued until the plant life time reaches its maximum. At this point in the calculation, if the desired number of trial runs have not been completed, the process is started again at time zero. As with any Monte Carlo process, confidence in the model prediction depends on the number of trial runs. This study used 100 trials per simulation to generate the time-dependent system failure probability estimates for each calculation where effective maintenance was studied.

Refurbishment is generally not a good model for a maintenance program. However, the purpose of this study is to demonstrate a technique for estimating system reliability characteristics when aging is controlled. In view of this goal, the simulation model can delay the renewal activity for a specified number of failures for an individual component. This time delay for renewal can be interpreted as establishing the periodicity at which the aging effects must be corrected in order to maintain a specified component failure probability, which, taken across the system, will control the system level unreliability.

6. AFW SIMULATION RESULTS

A simulation model of the Surry AFW was used to calculate time-dependent system unavailability for three different levels of T&M activity. The calculations performed for the study are defined as follows:

BC-GAO: This base case calculation uses the data from Table 1 and assumes that all testing or maintenance activities result in repair.

BC: This base case calculation uses the data from Table 1 and assumes that components are refurbished with respect to aging effects whenever the component fails. The failures are determined randomly based on the component's respective failure probability. Upon component failure, the component age scale is reset to zero, thus simulating periodic refurbishment of the components to address aging effects.

BC-D: This calculation uses the base case data but the maintenance activity is modified to allow a delay in the refurbishment of failed components. The components are randomly failed, but the aging scale is not reset at each time (as in BC). In this case, the maintenance activity for each component is allowed to be repaired until a specified number of failures have occurred, thus simulating periodic refurbishment at longer time intervals than those in BC.

The time-dependent failure probability calculated for the case BC-GAO is presented in Figure 3. From the figure, it can be seen that the system failure probability becomes large at extended plant life unless corrective maintenance is used to adequately address aging. Table 2 provides failure probabilities at selected system ages from the simulation. The data used to generate the curves presented in the figure indicate that from 1 yr (before aging becomes a significant contributor to failure probability) to 60 yr, BC-GAO experiences an increase in failure probability by a factor of 123. This increase is unacceptable and needs to be controlled by the maintenance program in order for the system to maintain an acceptable level of reliability.

Maintenance effectiveness is simulated by refurbishing the components at periodic intervals. The calculation BC, which has been designated as the base case, simulates component refurbishment and eliminates aging effects each time a component fails. This calculation sets a base refurbishment interval. The aging scales are reset and thus simulate an ideal

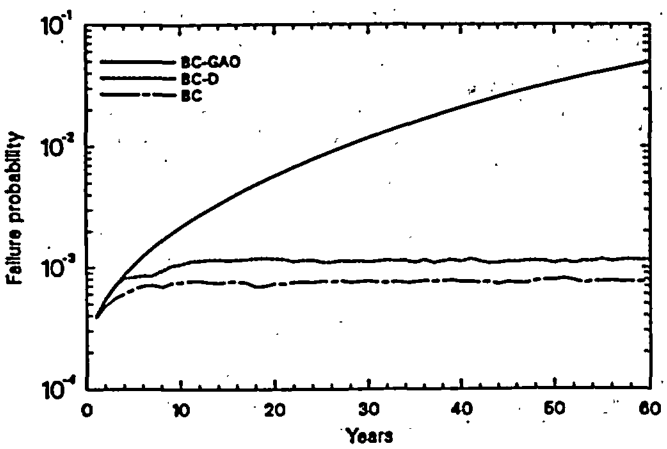


Figure 3. Relationship of system failure probability for noneffective versus effective maintenance practices.

TABLE 2. SELECTED SYSTEM FAILURE PROBABILITIES FOR SIMULATION BC-GAO

System Age (yr)	System Failure Probability
1	4.0E-4
30	1.2E-2
40	2.1E-2
60	4.9E-2

repair, but the reset does not take into account any preventive maintenance. The calculation BC-D demonstrates the sensitivity of the system failure probability to increases in the periodic refurbishment interval. In this calculation, maintenance is simulated by allowing the aging scale to reset after a specified number of failures have occurred. In the current calculations, the number of failures before aging reset occurred was determined arbitrarily for each aging component based on the base case results. For example, in the simulation BC, each air-operated valve experienced an average of approximately 23 failures in the 60-yr simulation. This relates to a failure every 2.6 yr on the average. In expectation of increasing the repair periodicity to over 10 yr, a reset value of 4 was chosen to allow aging reset to occur every fourth failure. The aging reset delay values used in the BC-D simulation and the average times between failure for both simulations are shown in Table 3. In actual practice, the choices of delay would be determined by the system reliability goals and engineering constraints dictated by plant operations.

Failure probabilities for the calculational cases with effective maintenance are presented in Figures 3 and 4. Comparison to case BC-GAO demonstrates the significant differences obtainable when effective maintenance is modeled. When maintenance is allowed to periodically correct the effects of aging, the shape of the curves indicate that essentially constant failure probability results from these activities. Depending upon the periodicity used in designing the maintenance activity or the magnitude of the aging rates, the system failure probability will attain differing levels of reliability. The fact that the curves are not smooth results from the use of a Monte Carlo process with a limited number (100) of simulations. However, sufficient simulations were made to assure that essential curve shapes and trends are present in the results.

TABLE 3. AVERAGE NUMBER OF YEARS BETWEEN REFURBISHMENT

Component/ Fault	BC Interval (yr)	BC-D Number Failures Before Refurbishment	BC-D Interval (yr)
AOV	2.6	4	6.1
STMBD	2.7	4	6.5
CKV	12.1	2	18.6
MDP-FR	NO[a]	10[b]	NO[a]
MDP-FS	5.6	4	15.2
TDP-FR	31.9	10[b]	NO[a]
TDP-FS	2.6	4	6.4

a. NO: Never occurred.

b. *It was expected that these would never occur or be* an insignificant contribution to the system failure probability. Therefore they were assigned large numbers.

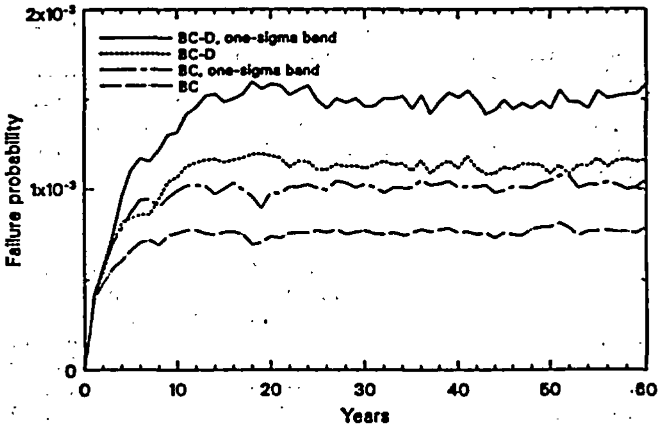


Figure 4. Display of effective maintenance simulations and simulation upper error bound.

The time-dependent failure probabilities for the curves BC and BC-D were fitted with a linear least squares regression algorithm to smooth the curve shape resulting from the nonconverged simulation runs, and estimate the essentially linear and time-independent value of failure probability for each simulation. Thirty years is the midpoint of the simulation time span and was chosen as representative for the estimation. Results of the regression analysis estimate the essentially constant failure probabilities to be 7.6E-4 for BC and 1.1E-3 for BC-D. Note that the failure probability at the end of the first year (before aging would become a significant contributor) was 4.0E-4.

Comparison of the linear regression results to the data presented in Table 2 demonstrates the significant impact maintenance effectiveness has on system failure probability. The system unreliabilities calculated for the base case and the sensitivity cases are essentially constant and significantly lower than the results for repair only. This indicates that system unreliability can be adequately controlled to be less than some goal level if a periodicity for aging control is established.

Since a Monte Carlo technique is used in the simulations, a measure of its uncertainty is needed to provide an indication of the accuracy of the results. The Monte Carlo statistics were calculated to estimate the convergence of the two simulations at each time-step. The results of the one-sigma upper bound for the effective maintenance simulations are displayed in Figure 4. These upper bounds represent an approximately 30% error bound for the 100 trial runs per simulation. The upper bounds could be decreased if more trial runs per simulation were performed.

For the system and aging rates under consideration, a study of the simulation results provide an indication of the periodicity with which the aging effects must be corrected to maintain the system failure probability at or below the calculated levels. Table 3 details the average time between component aging resets calculated by the simulation and reflects the periodicity of refurbishment for the component types. Examination of cases BC and BC-D indicate that while the average system failure probability increases by approximately 45% (7.6E-4 to 1.1E-3), the periodicity of necessary refurbishment increases by over a factor of two for all components except check valves. This result indicates that the system unreliability is not linearly related to the periodicity of aging effects correction, and a plant maintenance program would have significant latitude in developing the desired maintenance periodicity limits. In fact, the values of refurbishment periodicity calculated in the current study are of such time intervals that aging control can easily be accommodated by a plant maintenance program.

7. CONCLUSIONS

Periodic control of aging effects allows an essentially constant system reliability to be maintained. The necessary system failure probabilities can be developed from the use of PRA techniques, and the techniques presented here can then be used to establish the necessary periodic control that will allow the system unreliability to be maintained below the desired level. This analysis does not take credit for preventative maintenance programs and the effectiveness of the repairs currently performed on components at operating plants. However, the procedures outlined in the paper could be coupled with a rule based system in which the effectiveness of maintenance is evaluated and used to modify the component aging rates. This combination would develop an analytical tool that would allow system unreliability to be monitored on a periodic basis to assure that the system reliability goals are being met. Thus, allowing plant management to balance refurbishment and replacement of components with overall system reliability.

8. REFERENCES

1. Bertucio, R. C., et al., Analysis of Core Damage Frequency from Internal Events: Surry, Unit 1, NUREG/CR-4550 Vol. 3, November 1986.

2. Vesely, W. E., Risk Evaluations of Aging Phenomena: The Linear Aging Reliability Model and Its Extensions, NUREG/CR-4769, April 1987.

3. Kalbfleisch, J. D. and Prentice, R. L., The Statistical Analysis of Failure Time Data, New York: Wiley, 1980.

4. Vesely, W. E., ltr to Satterwhite, D. G., Status Report Package for the Aging Subcontract, INEL, NRC FIN-A6389, March 1987.

5. Meale, B. M. and Satterwhite, D. G., An Aging Failure Survey of Light Water Reactor Safety Systems and Components, NUREG/CR-4747 Vol. 2, July 1988.

NOTICE

This paper was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this paper, or represents that its use by such third party would not infringe privately owned rights. The views expressed in this paper are not necessarily those of the U.S. Nuclear Regulatory Commission.

CONDITION MONITORING OF ACTIVE COMPONENTS IN NUCLEAR POWER PLANT

SHIGERO MASAMORI
MITSUBISHI HEAVY INDUSTRIES, LTD.
KOBE SHIPYARD & MACHINERY WORKS

CHIAKI YASUDA
MITSUBISHI HEAVY INDUSTRIES, LTD.
TAKASAGO R & D CENTER

TAKESHI SATOH
MITSUBISHI HEAVY INDUSTRIES, LTD.
KOBE SHIPYARD & MACHINERY WORKS

SUMMARY

In Japan, average PWR plant capacity factor has increased step by step. The age-related degradation, however, has to be taken into account in maintenance activities to maintain plant capacity factor and extend plant life. Especially condition monitoring and diagnosis of active components, therefore, has to play a great role to ensure nuclear safety and maintain productivity of entire facility. This paper provide the R & D status of two types of condition monitoring techniques of active components.

I. Introduction

In Japan, average PWR plant capacity factor has increased step by step --- from 54.1% in 1978 to 78.4% in 1985. The age-related degradation, however, has to be taken into account in the maintenance activities to maintain plant capacity factor. Accumulated information and knowledge obtained through maintenance activities are contributing to develop more optimized preventive maintenance program and to stimulate new maintenance technology development such as robotics, incipient failure detection and machine condition monitoring.

This paper provide not only general status of nuclear maintenance in Japan but also the research and development status of the following diagnosis and monitoring techniques of active components as the upgrade preventive maintenance.

(1) diagnosis expert system for rotating equipment
(2) diagnosis expert system for PWR feedwater system

II. General Status of Nuclear Maintenance in Japan

A. Historical trend

Maintenance programs are becoming more complex in general trends. This increasing complexity has caused utilities to become increasingly concerned with more optimized maintenance management.

B. Maintenance strategy

Preventive maintenance is the basic strategy followed as extensive as reasonably achievable. In light of public acceptance and social impacts, highest priority for maintenance is given to preclude any type of incident and/or failure which will result in forced shutdown and radioactive material release to public environment. A continuous program throughout the year is applied to identify components that could cause potential availability problems. The preventive maintenance program is then focused on them.

C. Optimization of scheduled annual maintenance activities

Because of resource requirements, engineering and technology applied and impacts on plant safety and performance, utility customer pays great attentions and considerations to scheduled annual maintenance activities. Extensive developmental and field efforts are provided by nuclear industry to reduce outage time and man-rem.

III. Diagnosis and Monitoring Techniques

Because nuclear plant components have been operated under the environment of high temperature, high pressure and high radiation, there supposed to be not only material degradation of pressure retaining components but also decrease of efficiency of active components such as rotating equipments.

Diagnosis and monitoring techniques will, therefore, be required to predict the above degradation.

A. Diagnosis and monitoring as ISI

Diagnosis and monitoring as ISI consists of the inspection required by technical code and the inspection planned by utility as the preventive maintenance. The main diagnosis and monitoring are as follows.

(1) Scheduled overhaul to inspect the degradation of components
(2) Non-destructive examination and leak test of class 1 - 4 components required by technical code (JEAC-4205)
(3) Functional test of safety-related system and components

B. Diagnosis and monitoring during plant operation

Diagnosis and monitoring during plant operation consists of daily patrol of components, surveillance of operation parameters, diagnosis by analyzing the trend of operation data and the state of art diagnosis.
The upgrade preventive maintenance requires especially the following state of art diagnosis.

(1) Loose parts monitoring system to detect the loose part in the primary loop.
(2) Diagnosis expert system for rotating equipments to detect and diagnose the vibration of pumps and fans by using expert system.
(3) Steam generator secondary side water chemistry on line monitoring system to monitor, analyze and diagnose the secondary water chemistry automatically and continuously.
(4) Fatigue monitoring system to calculate the actual cumulative usage factor by means of measuring temperature of the outer surface of component.
(5) Crack monitoring system to detect the crack of pressure retaining components by non-destructive technique and to predict the remaining life by ASME SEC XI method.
(6) Diagnosis expert system for PWR feedwater system to identify defective components and to provide specific guide line for repair.

IV. Diagnosis Expert System for Rotating Equipment

A. Techniques to be developed

Nuclear electric generating power plants are equipped with a number of rotating machines which play an important role in the plant system. Therefore, respective users make an effort to improve machine reliability and availability by cooperating with the plant manufacturers. Especially, the maintenance of such equipment for healthy operation as well as the improvement of the prevention technology against plant failures have become essential requirements. As is well known, various types of sensors have been installed to monitor the condition of the main rotating machines in the plant system, such as main steam turbines, for safe operation and for a satisfactory maintenance of the equipment.

The vibration data is monitored at the rotating machines as well as the various process data such as temperature, pressure, flow rate and so on, because the vibration data is a quite effective indicator of the rotating machinery health and is sensitive to the failure and malfunction of the rotating machines.

However, as the conventional vibration supervisory system monitors only the overall level of the vibration, it can detect the machine failures, but cannot detect them at the incipient stage. Further, as the conventional system can only plot the data on a chart recorder, it is impossible to estimate the cause of the failure after the rotating machine stops due to the failure. Therefore, an advanced supervisory system has been developed recently.

This system can not only monitor the condition of rotating machines continuously, but also detect failures at the incipient stage. Furthermore, this system can automatically estimate the cause of the failure by using the conventional automatic diagnostic method. There are two types of the advanced supervisory system.

One is the system based on the mini-computer, which can monitor the condition of many rotating machines in the nuclear power plant, such as the main steam turbine generator, reactor coolant pumps, feed water pump turbines and so on. Figure 1 shows the overview of this system and Figure 2 shows the hardware configuration of this system.

Another is the system based on the micro computer, which can monitor the condition of several auxiliary rotating machines in the power plant, such as reactor vessel cooling fans, containment vessel cooling fans and so on.

Figure 3 shows the overview of this system and Figure 4 shows the hardware configuration.

These two systems are named Machinery Health Monitoring System: M.H.M. system. The main functions of these two systems are the followings;

(1) Data Analysis and Data Acquisition Functions
(2) Data Display and Data Report Function
(3) Alarm Function for Machine Failure
(4) Diagnosis Assist Function
(5) Balance Weight Calculation Function

The last function is available for only the mini-computer based system. The multiprograms are simultaneously executed in these systems in order to realize the main functions mentioned above and to make it operationally easy. Therefore, the processing flow is quite complicated. Figure 5 is simplified processing flow of these systems. As shown in Figure 5, the process can be classified according to the continuous monitoring process and diagnostic process for machine failure. The former process is always executing while this system is on power. The latter process is executed when the system detect the machine failure.

The alarm function is composed of three kinds of inspections as follows.

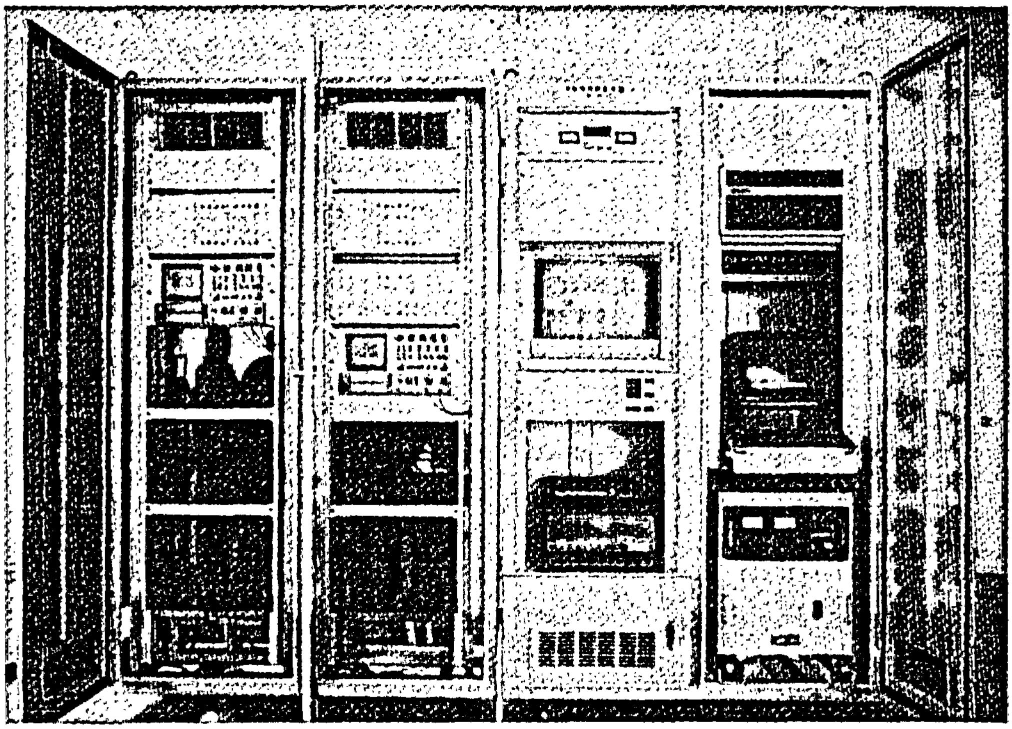


Fig. 1 Overview of Machinery Health Monitoring System based on Mini-computer

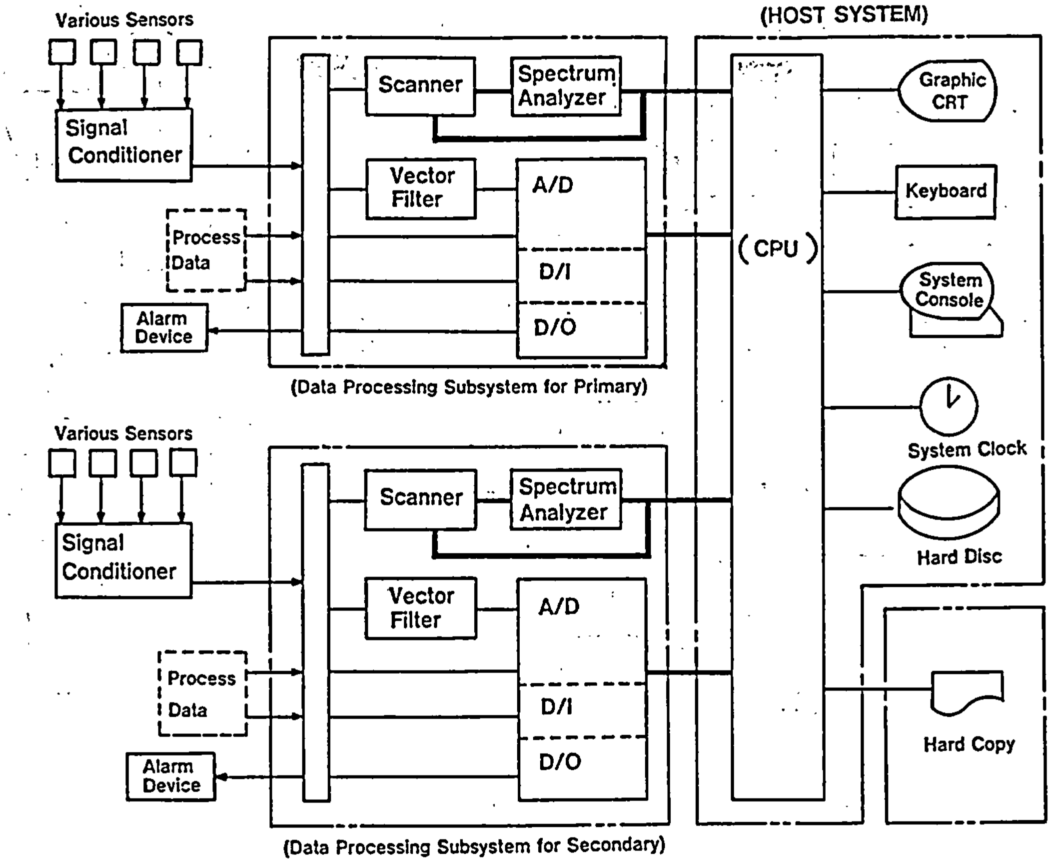


Fig. 2 Hardware Configuration of Machinery Health Monitoring System based on Mini-computer

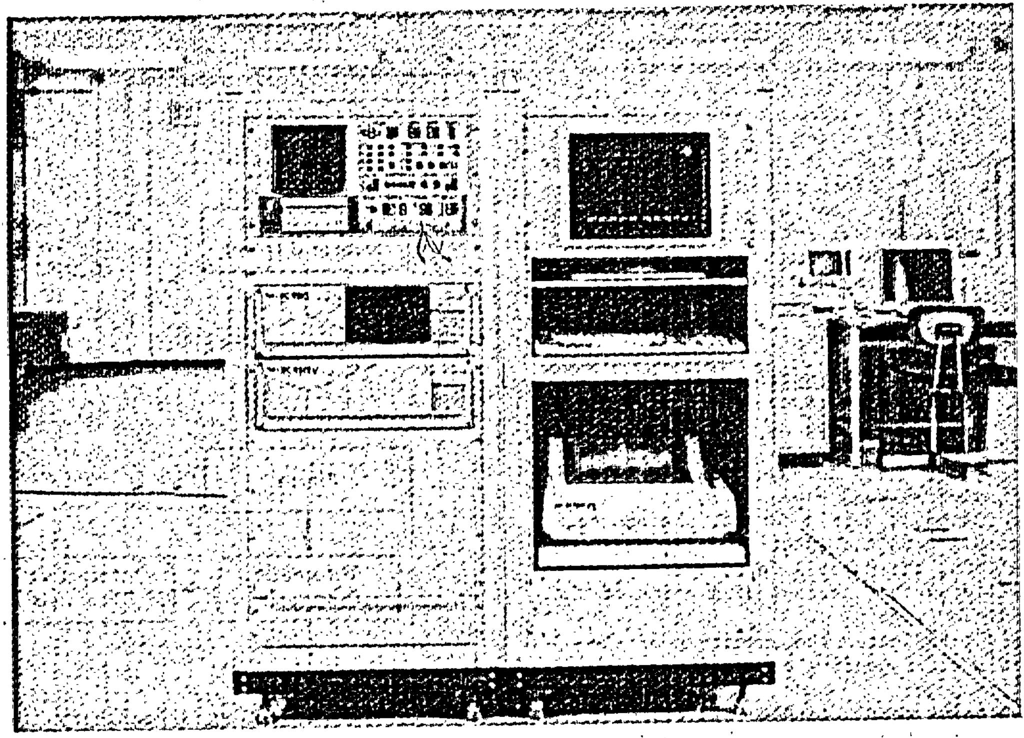


Fig. 3 Overview of Machinery Health Monitoring System based on Micro-computer

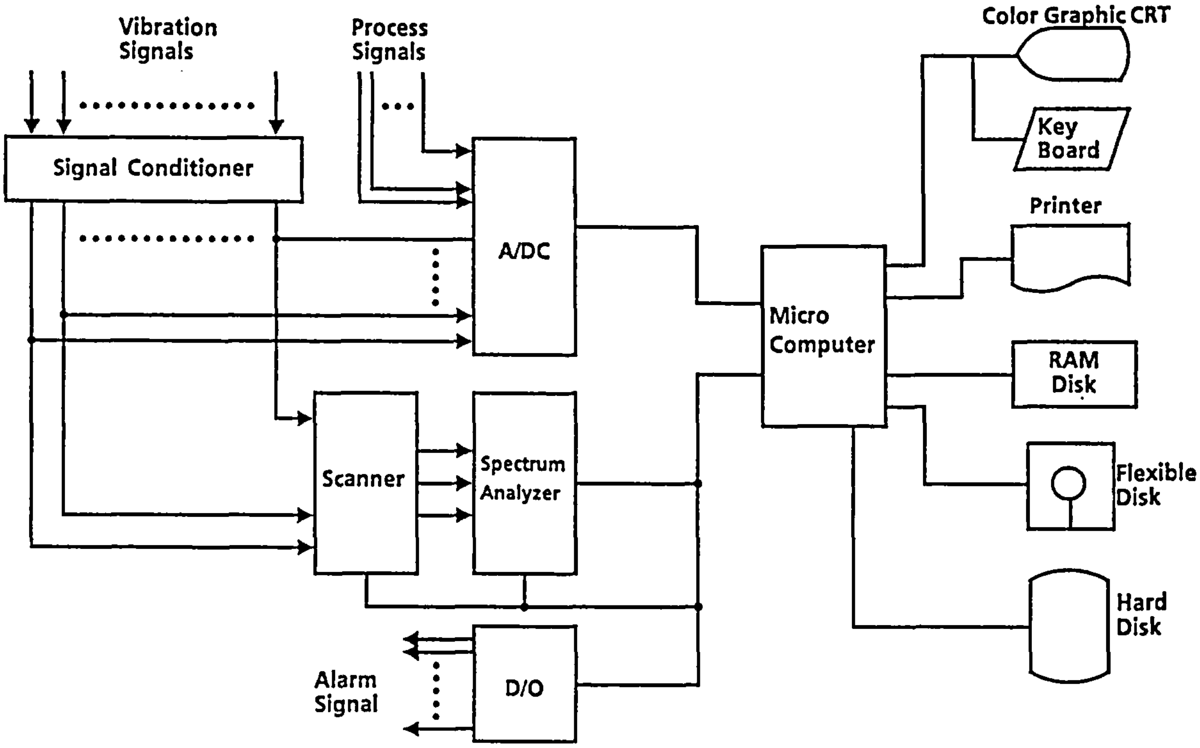


Fig. 4 Hardware Configuration of Machinery Health Monitoring System based on Micro-computer

(1) Vibration Level and Vibration Vector
(2) Vibration Level Trend and Vibration Vector Change
(3) Frequency Component based on Vibration Spectrum

Due to the above inspection, these systems can detect the machine failure at the incipient stage. Diagnostic assisted function estimates the cause of machine failure, when machine failure occurs and is detected. This function utilizes the weighted point matrix method. This method estimates the possible causes of machine failure by using the causal matrix shown in Table 1 is an example of the causal matrix for general rotating machines. Each row of the causal matrix corresponds to each cause of machine failure which is possible for a general rotating machine.

On the other hand, each column corresponds to an observed system. The symptoms are the various analyzed vibration data, mainly the frequency component data analyzed by the spectrum analyzer and the vector filter.

Each element of the causal matrix indicates the intensity of the relationship between the corresponding cause and the corresponding symptom. For example, it indicates in Table 1 that the rotative speed vibration component increase if a large or significant unbalance exists in the rotating machine.

In the weighted point matrix method, the scores, which are decided based on the spectrum data, the vibration vector data, the process data, and the design data, are multiplied by the corresponding point of the causal matrix and summarized.

The final score indicates the magnitude of the possibility for the corresponding cause of the machine failure.

The weighted point matrix method work well for the simple problem in which the number of the cause is small and the symptom is not complicated. However, it has the following problems;

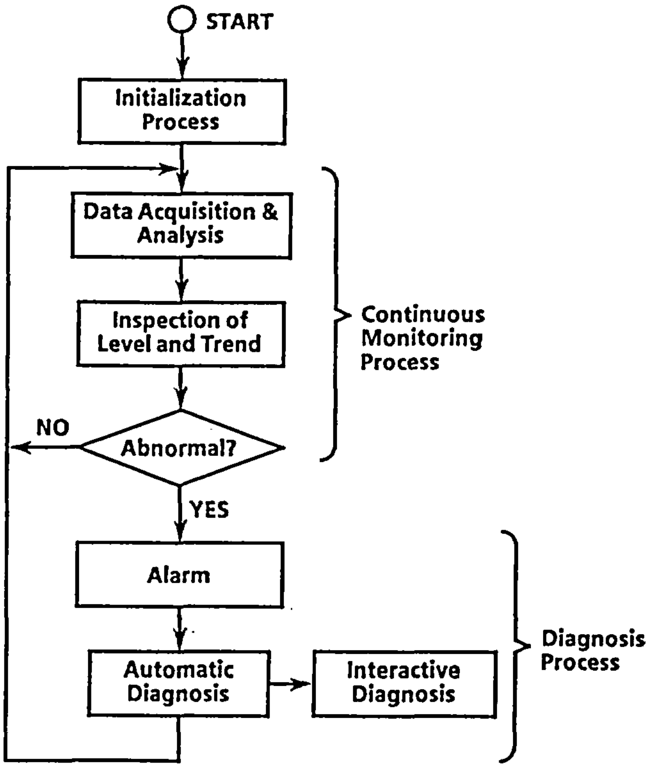


Fig. 5 Processing Flow of Machinery Health Monitoring System

Table 1 Example of Causal Matrix

Causes \ Phenomena	Low Cycle Vibration	0.3 ∫ fn 0.49	1/2fn	0.51 ∫ fn 0.99	Rotating Compone-nt	2fn	3fn	Higher fn	fnZ	1st Critical speed fc1	2nd fc2 3rd fc3	Gear Mesh Frequency fG	2fG	3fG	Acoustic Frequency
① Roller Bearing Defect															* * *
② Rub	*	*	*	*	*					*	*	*			* * *
③ Cracked Rotor					* * *	* * *	*	*							*
④ Cavitation															* * *
⑤ Damaged Gear															* *
⑥ Electric Magnetic Vibration															* * *
⑦ Blade Vibration									* * *						
⑧ Mis-alignment					* *	* *	*	*							
⑨ Asymmetrical Rotor						* * *									
⑩ Unbalance					* * *										
⑪ Bending Shaft					* * *										
⑫ Draft Core		* * *													*
⑬ Gap and Nonlinearity		*	* *	*	*	*	*	*							
⑭ Oil Whip		* * *								* * *					
⑮ Steam Whirl		* * *	*	* * *						* * *					
⑯ Surging	* * *														

(1) Not applicable for large scale problem
(2) Difficulty of causal matrix construction
(3) Difficulty of algorithm editing
(4) Difficulty of design data handling
(5) Difficulty of non-numerical information

One of the solution for the above-mentioned problems is the "Expert System."

B. Concept of diagnosis expert system for rotating system

In order to improve the diagnosis performance, two projects, concerning diagnosis system for rotating equipments, are being performed.

(1) Interactive diagnosis system for the various small size machine
(2) On-line diagnosis system for the large size machine

The former is developed for the various small size rotating equipments such as various auxiliary pumps, fans and so on. In general, the cost of the instrumentations, such as various vibration sensors, signal conditioners and wiring installation, is more expensive than the cost of the computer system. Therefore, this system utilizes the portable data acquisition system that is Snapshot made by Bently, Nevada. Although this portable data acquisition system is very compact and very light, it can simultaneously acquire two vibration signal waves with keyphase. In addition, it can memorize many data and transfer data to the computer through the IEEE interface bus. Therefore, maintenance personnel can easily handle it to acquire vibration data of many rotating equipments which are set up here and there in the power plant. Namely, the interactive diagnosis system is much cheaper than the on-line diagnosis system.

The interactive diagnosis system is shown in Figure 6. In this system, many types of knowledge base and rule base are necessary because this system can diagnose various rotating equipment. In addition, as only vibration data is automatically acquired, the operational mode must be interactive in order to acquire another data such as the process data, design data and so on.

Fig. 6 Overview of Interactive Diagnosis Expert System

The latter system is developed for large size rotating equipments such as turbine, reactor coolant pump and so on. This system can continuously acquire various data, such as vibration, process and so on. Therefore, this system can quickly respond to machine failure. For example, if the failure happens, this system can quickly and exactly estimate the cause of the machine failure.

Figure 7 shows the hardware configuration, and Figure 8 shows the software configuration. As shown in Figure 9, this system is composed with the data acquisition subsystem and the diagnosis subsystem. This is the reason why the processing efficiency is high.

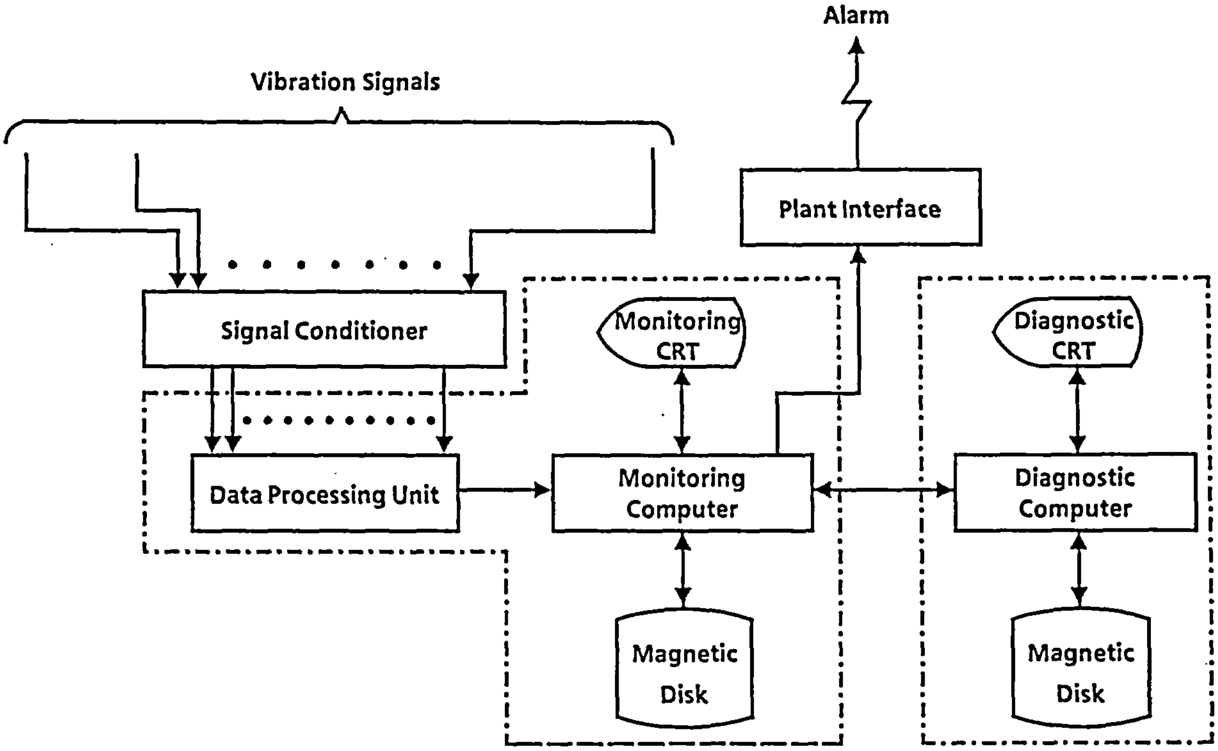


Fig. 7 Hardware Configuration of On-line Diagnosis Expert System

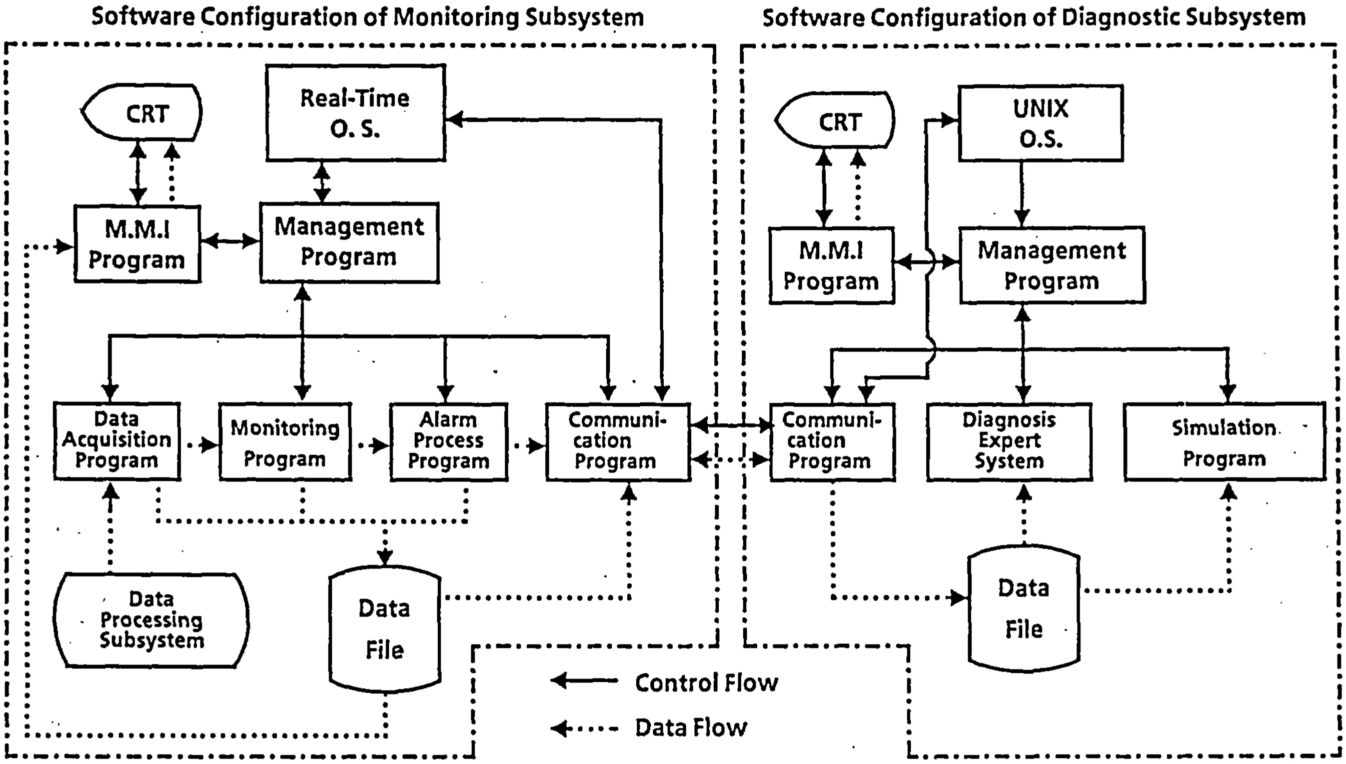


Fig. 8 Software Configuration of On-line Diagnosis Expert System

The example of diagnosis results by the proto-type system are shown in Figure 9. This example is for reactor coolant pump. Figure 9 shows the display of the final diagnosis result by using the design rule base, analytical data rule base, and experienced knowledge base in addition to the on-line data rule base.

On-line data rule base is based on the vibration and process data monitored on-line. The design rule base is based on the design data such as the natural frequency of shaft and the number of impeller. The simulation data rule base is based on the simulated data by simulator program or analyzed data by analysis program, such as the vibration vector estimated by unbalance response simulator and the correlation between the vibration data and the process data.

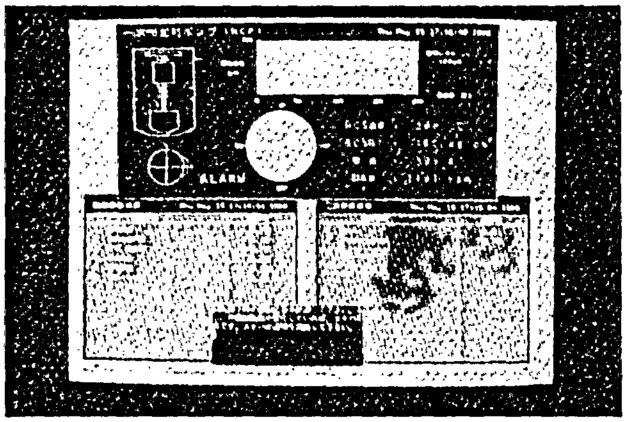


Fig. 9 Example of Display of Proto-type On-line Expert System

C. Development milestone of diagnosis expert system for rotating equipments

The interactive diagnosis expert system for auxiliary pumps and fans in nuclear plant will be developed according to the following steps;

(1) Design of data acquisition system
(2) Building acquisition system
(3) Acquiring and analyzing actual data in the plant
(4) Construction of rule base based on vibration data
(5) Construction of rule base based on design data
(6) Design of expert system
(7) Building proto-type system
(8) Trial use of proto-type system
(9) Improvement of proto-type system
(10) Actual use of system

The schedule of each development step is shown in Table 2. As shown in Table 2, the data acquisition system has been developed already. Therefore, the data acquisition phase will be started soon.

The on-line diagnosis expert system for reactor coolant pump will be developed according to the following step;

(1) Design of system concept
(2) Construction of fault tree for rotating machine
(3) Design of data acquisition system
(4) Construction of rule base based on on-line data
(5) Construction of rule base based on design data
(6) Construction of rule base based on analyzed data
(7) Design of simulation system for diagnosis
(8) Building proto-type system
(9) Trial use of proto-type system
(10) Improvement of proto-type system
(11) Actual use of improved system

The development schedule of on-line diagnosis system is shown in Table 3.

Table 2 Development Schedule of Interactive Diagnosis Expert System

#	Items	F.Y. '87	F.Y. '88	F.Y. '89	F.Y. '90
1	Design of Data Aquisition System	▭	▭		
2	Building Aquisition System		▭		
3	Aquiring and Analyzing Data		▭	▭	
4	Rule base based on Vibration Data		▭	▭	
5	Rule base based on Design Data		▭	▭	
6	Design of Expert System		▭	▭	
7	Building Proto-type System			▭	
8	Trial Use of Proto-type System				
9	Improvement of Proto-type System				▭
10	Actual Use in Plant				▭

Table 3 Development Schedule of On-line Diagnosis Expert System

#	Items	F.Y. '87	F.Y. '88	F.Y. '89	F.Y. '90
1	Design of System Concept	▭			
2	Construction of Fault Tree	▭	▭		
3	Design of Data Acquisition System	▭	▭		
4	Rule base based on On-Line Data	▭	▭		
5	Rule-base based on Design Data	▭	▭		
6	Rule-base based on Simulation Data	▭	▭		
7	Design of Simulation System		▭		
8	Building Proto-type System		▭	▭	
9	Trial Use of Proto-type			▭	▭
10	Improvement of Proto-type			▭	▭
11	Actual Use in Plant				▭

V. Development of Diagnostic Expert System for PWR Feedwater System

A. Techniques to be developed

Maintenance activities to be conducted at plant field consists of identification of defective component, repair, inspection and adjustment. From the human factor point of view, there are still rooms for improvement in present maintenance practice for Feedwater System such as listed below.

(1) Cause inferences depend on experience and knowledge of assigned plant personnel mostly.
(2) Number of experienced maintenance crew is decreasing and limited technology and knowledge transfer among the organization.
(3) Difficulty to classify the significance of off-normality.
(4) Limited use of process data for diagnosis.
(5) Difficulty to collect important data for diagnosis such as pressure of air components.

In order to resolve maintenance related human factor issue, computer aided expert system which supports crew and shortens maintenance time has been developed.

B. Concept of diagnosis expert system for PWR feedwater system

1. Design functions Identification of defective components and indication of maintenance guidance is performed for supporting maintenance crew when off-normality of Feedwater System occurs during plant operation.

2. Coverage of components diagnosed As illustrated in Fig.10, system diagnosis covers essential components such as Feedwater Valves (main components of Feedwater System), Feedwater Pumps, air components actuating valves, analog control components and sensors.

3. Diagnostic technique Knowledge engineering technique in conjunction with specialists' knowledge of experienced maintenance crew is applied to this system. In order to minimize uncertainty and to improve reliability of diagnosis, simulation model based on plant physical parameters are also incorporated.

4. Two phase diagnosis Diagnosis of this system is classified into two phases to realize efficient diagnosis. In the first phase diagnosis, defective component is identified based on the data derived from instrument rack. When the first phase diagnosis fails in identification of defective component or needed to go detailed diagnosis, the second phase diagnosis is initiated.

The second phase diagnosis is processed mainly based on man-machine dialogue with feeding additional field data if necessary.

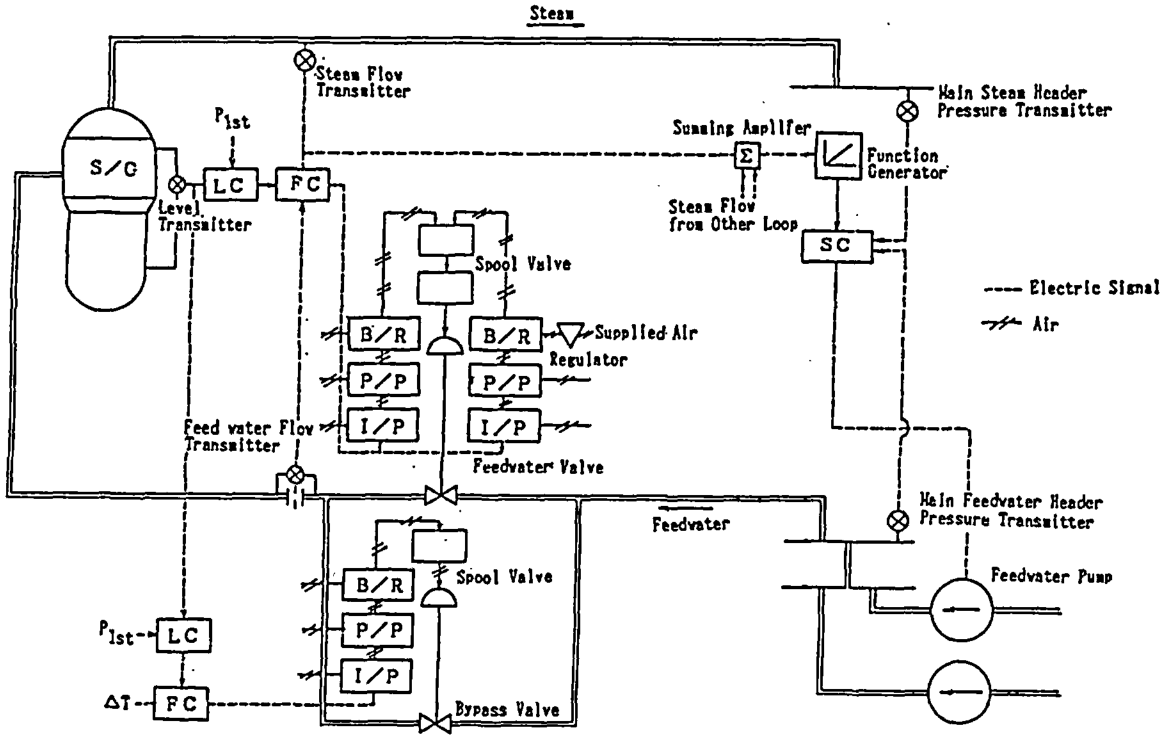


Fig. 10 Coverage of Components Diagnosed

5. System configuration System software configuration is illustrated in Fig.11 and roles of each program are as follows.

(1) Dialogue management program - This program performs input and output management of this system. (Data input to diagnostic program, man-machine dialogue with maintenance crew, display of maintenance guidance etc.)

(2) Diagnostic program - This program is a inference engine with identify defective components and specify maintenance guidance using knowledge data base and data requested along with dialogue management program.

(3) Knowledge data base - Knowledge data base contains available knowledge of experienced maintenance crew, and design information which are needed to identify defective components.

C. Development of prototype system

To realize system concept discussed in previous section, prototype system software FORMAS-FWS (Frame Oriented Maintenance Assistant System for PWR Plant Feed Water System) has been developed.[1]

1. Knowledge source Classified technical problem, causes and countermeasures by symptoms and components wise were obtained and package of knowledge was developed. Potential candidates of off-normal components, process of explication of cause and maintenance guidance are also obtained by literature survey.

Design information, component dynamic response characteristics to be used for simulation model are also gathered.

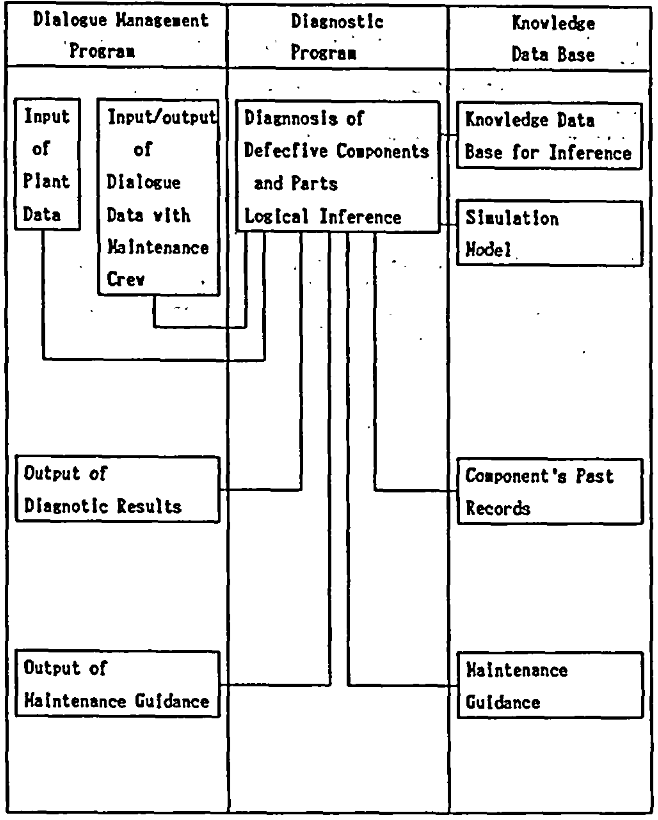


Fig. 11 System Configuration

2. Basic approach to diagnostic knowledge development Offnormal symptoms are determined basically by Feedwater Flow information. These symptoms are categorized into four(4) phenomena, namely offnormal increase, decrease, lock and vibration of Feedwater Flow. Diagnosis is proceeded mainly by composed knowledges of components consisting feedwater system, under the assumption that physically defective components certainly exist when three phenomena, increase, decrease or lock of Feedwater Flow occur. Namely, several components are handled as one complex component and verification of this complex component is performed. If this complex component is verified to be offnormal, verification of each component offnormality is performed respectively.

As mentioned above, using manner of diagnosis addressing from wide range of components complex to smaller one, suitable judgement can be made and missing of defective components can be prevented.

Vibration of Feedwater Flow is caused not only by individual component but also by combination of process operating parameter. Therefore, main diagnostic knowledge for assumed phenomenon is composed by experience of skilled maintenance crew and sub diagnostic knowledge is composed by design information of Feedwater system. During verification sequence of component offnormality, the component which experienced higher likelihood of offnormal in the past is prioritized. Certainty of cause verification of defective component is determined by certainty factor. This factor has positive value which valued based on experience.

3. Composition of diagnostic knowledge Diagnostic knowledge package consists of following five frames.

(1) Input data treatment frame - Plant data calculation and pre-processing are performed in this frame to be used in other frames.
(2) Question frame - To obtain necessary data for diagnosis, this frame makes questions to maintenance crew. Questions are asked sequentially according to diagnosis flow and inconsistent and/or inlogistic question can be avoided.
(3) Offnormality frame - This frame is defined on each offnormal symptom and describes evaluation methods of symptoms and candidates of defective component in case that symptom occurs. This frame plays important role for increasing diagnosis efficiency.
(4) Cause frame - This frame is defined for each defective component and each intermediate cause. This knowledge consists of condition to verify defective component, rules for determining certainty of verification, methods for targeting defective components and maintenance method for defective components. Therefore, this frame plays main role for diagnosis.
(5) Maintenance frame - This frame determines optimum maintenance method from candidates of maintenance guidance for verified defective components.

Adapting above composition of frames accumulation of knowledge, analysis of content of knowledge and efficient diagnosis can be executed.

4. Reasoning control As illustrated in Fig.12, reasoning control flow of FORMAS-FWS consists of first phase diagnosis (from data input to explanation function) and second phase diagnosis (from cause diagnosis to explanation function.) Defective components are identified mainly using data derived from instrument rack in first phase diagnosis and defective components are identified by dialogue data with maintenance crew following first phase diagnosis. These data are managed as follows.

(1) Data input - First of all dialogue data and plant data are input in this system and pre-processed.
(2) Comprehension of offnormal system and status - Using pre-processed data, offnormal symptoms are selected by evaluation methods which are described in Abnormality Frame.
(3) Determination candidate of offnormal component - Candidate components (cause frames) which are considered as causes of offnormal event determined from offnormality frame.
(4) Cause diagnosis - Offnormality verification condition which are described in pointouted cause frame is evaluated by inputting necessary data and its results are indicated as certainty factor. This value is an index of degree of offnormality verification of candidate components. And if maintenance frame is available, these frames are assigned as candidates for maintenance guidance.
(5) Determination of countermeasures - Appropriate maintenance guidance is determined by evaluating conditions in maintenance frame.
(6) Explanation function - This function explains status of above processing in diagnosis flow to maintenance crew and explains the reason why these results extracted.

5. Output display This system displays processing status, abnormal symptoms occurred, identified defective components and maintenance guidance etc. in multi-window of CRT.

6. System specification

Calculator ... Micro VAX II
Language ... VAX LISP, FORTRAN

Number of Input Data Management	... 17
Number of Question Frame	... 43
Number of Offnormality Frame	... 3
Number of Cause Frame	... 50
Number of Maintenance Frame	... 34

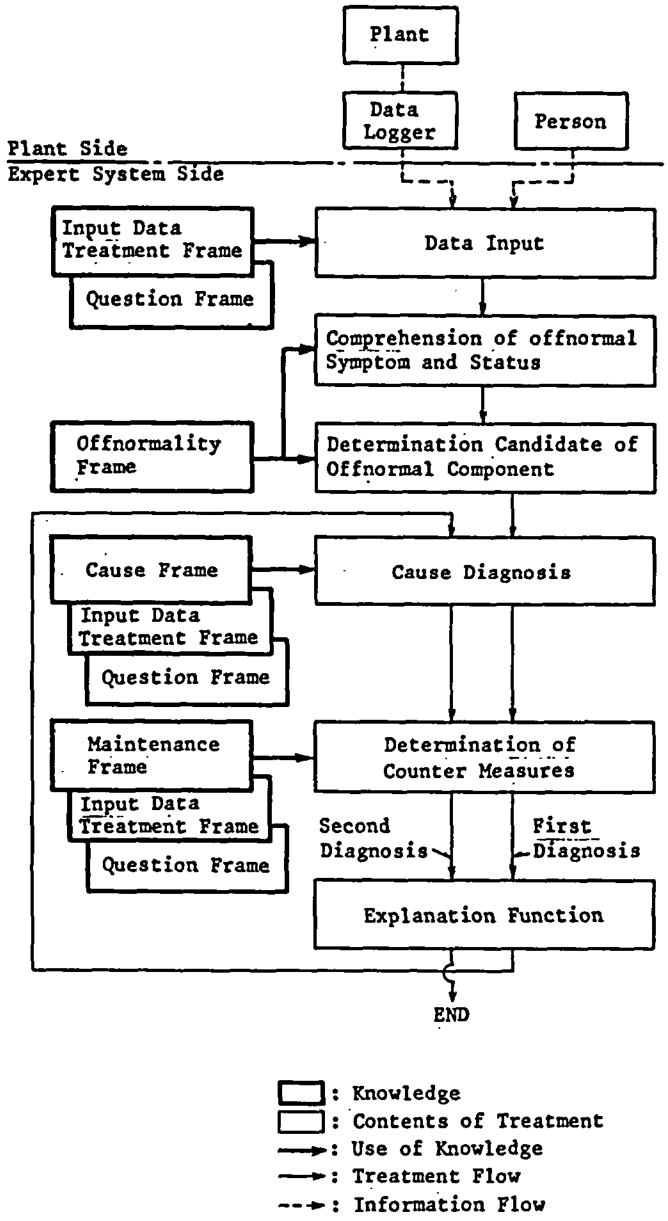


Fig. 12 Reasoning Control Flow

7. Evaluation Following evaluations are obtained from operating this prototype system.

(1) Defective component can be flexibility identified corresponding to offnormal event of Feedwater system and proper maintenance guidance can be provided.
(2) It is useful to perform step wise diagnosis into two phases.
(3) Diagnosis is successfully executed within a time frame of second.

VI. Conclusion

We believe that condition monitoring systems of active component which are now being developed are very useful and effective for plant preventive maintenance. And therefore, we intend to develop these condition monitoring systems for actual use.

Reference

(1) OKAMACHI, M., MURATA, R., ABE, H., SATO, T., "Development of diagnostic expert system for PWR plant Feedwater System," Lecture papers, Japan Society of Mechanical Engineers, No.870-1 (1987) 37-40.

MAINTENANCE MANAGEMENT OF NUCLEAR POWER PLANT IN JAPAN

- PRESENT SITUATION OF PREVENTIVE MAINTENANCE -

TAKUYA HATTORI
TOKYO ELECTRIC POWER CO.

Introduction

In Japan, more than 20 years have passed since the first commercial nuclear power plant started operation and now, as of June 1988, total 35 plants are in operation. (total 27,881MWe, BWR 18 plants/ 15,117MWe, PWR 16 plants/12,598MWe, and GCR 1 plant/ 166MWe)

Tokyo Electric Power Co. (TEPCO), the biggest utility in Japan, has 11 BWR operating plants (10,196 MWe) which shares about 36% (67.1 Billion Kwh) of our total electricity generation in 1987 FY.

In past 5 years, performance indicator has been steadily improved year by year.

In 1987 FY, an average capacity factor of 11 plants is 76.4%, an average scram rate due to an incident is 0.1 times/reactor.year, an average radiation exposure is 355 man-rem/reactor.year and these performance indicators are further improved in 4 plants which belong to the latest generation.

We have systematically implemented a modification and improvement of plant facility and a precise maintenance management, and especially promoted a complete preventive maintenance as well.

We believe that a stable plant operating result is successfully achieved by our continuous efforts described above.

I would like to introduce hereinafter a present situation of maintenance management in nuclear power plants in Japan, based on the cases in TEPCO.

I. Basic Conception of Maintenance Management

The fundamental policy for securing safety of nuclear power generation in Japan is to give the highest priority to the prevention of an abnormal event. We are taking concrete measures to insure proper maintenance management because precise maintenance management needs to be maintained before safe plant operation can be achieved. Maintaining sound condition of plant equipment is accomplished by:

1st: Voluntary maintenance management by electric utility
2nd: Strict quality control
3rd: Completeness of preventive maintenance

These principles are closely interrelated and their basic concept is deeply rooted in Japanese social culture.

(1) Voluntary maintenance management by electric utility

Japanese law specifies that the electric utility must maintain and manage the facilities in nuclear power stations so as to keep an appropriate quality level, and confirm the functional operation of systems and their reliability by periodically performing equipment disassembly and inspection. For that purpose, we shutdown the plant for about 3 months after about one year of operation and we perform a periodical inspection in order to take complete measures for preventive maintenance of the equipment in conjunction with plant refueling. However, this maintenance and inspection required by the law is only a part of our whole preventive maintenance program. The plant reliability is precisely maintained in principle by the conscious and voluntary effort of the electric utility, in other words, the major premise to maintain the plant reliability is to carry out the precise plant maintenance under the voluntary management by the utility.

(2) Strict quality control

In order to secure stable plant operation and prevent the occurrences of abnormal events, it is most important to build highly reliable facilities by using high technology and complete quality control at each stage of design, manufacturing, construction and pre-operation.

After start of commercial operation we execute a complete preventive maintenance program to maintain the high quality level achieved during the plant construction period. The core of the program is strict quality control.

We make every effort to secure and improve quality by adopting a comprehensive program, so called "TQC method." Narrow range and passive quality control methods applied only to prevent the production of inferior goods are not the case in Japan. The Total Quality Control or TQC method covers a wider range of fields such as plant facility, fieldwork, worker and management in pursuit of the common target of securing stable plant operation, and everyone from a field worker to a head office management being involved in the TQC method so as to improve quality.

(3) Completeness of preventive maintenance

a) Based on our fundamental policy to give the highest priority to the prevention of abnormal events, we try to carry out preventive maintenance as much as possible, rather than corrective maintenance. As a result, we have accomplished high plant availability.

Though it is difficult to make a quantitative evaluation on the effect on reliability improvement by means of preventive maintenance, we believe that preventive maintenance is well worth the cost based on public reliance and support obtained by the continuation of safe and stable plant operation.

However, the preventive maintenance has some problems, as follows:

* Maintenance costs and manpower requirements can be high.

* There is a possibility of causing trouble during maintenance work due to a human error.

* It is mainly effected on the limited trouble caused by deterioration and wear.

b) Surveillance tests and preventive maintenance

Surveillance tests are very important for the following reasons:

* The deteriorated parts, which need repair, can be identified during plant operation.

- Soundness of a whole system can be confirmed by surveillance testing while it is difficult to confirm this solely by means of maintenance of parts and components.

However, surveillance testing can not substitute for preventive maintenance because the surveillance test itself does not contribute to the quality improvement of equipment. From a view point of securing quality of equipment, the surveillance test is defined as a supplement to equipment maintenance and also as passive preventive maintenance.

II. Present Situation of Preventive Maintenance and Normal Maintenance

II-1 Maintenance management during plant operation

Inspection and maintenance are carried out not only during a periodical outage, but during a plant operation as well. In order to secure stable plant operation, it is absolutely important to take an appropriate countermeasure in advance against any potential problems that may happen during plant operation. It is necessary to identify equipment condition during plant operation, so that we can make a precise inspection plan during the outage. Therefore, the staff of the plant maintenance contractor in charge of actual outage work stays on site permanently and keep watch on the conditions of plant operation and maintenance.

(1) Types of Inspection

The inspection and repair which is carried out during operation is as follows:

a. Patrol and monitor by an operator and maintenance staff

b. Regular inspection and calibration by a maintenance staff

c. Regular test and function test (surveillance test)

d. Repair of failures (MRF) found as a result of above a, b, and c. (MRF: Maintenance Request Form)

(2) Observation organization

* Regular patrol by a shift operator	3 times/day
* Observation via TV monitor by shift operator	as required
* Intensive monitoring of instrumentation by maintenance staff	once/month
* Patrol by a special team of plant management staff	twice/month
* Patrol by a plant manufacturer's staff stationed at the site	once/week

In addition to the above regular monitoring, we execute a special patrol and inspection with contractors and confirm if there is any abnormal event before the year-end and summer time when electric demand reaches peak, and make every effort to secure reliability.

(3) Result of MRF Analysis

Equipment problems that occur during plant operation are quickly repaired without plant shutdown by means of the MRF submitted by a shift operator to the maintenance department, excluding the cases when a repair can't be made without plant shutdown or it takes too long to procure the parts needed for repair or replacement.

The number of MRF issuances and contents of MRF differs at each plant, and in case of Fukushima Daini unit No.2 (2F-2), our representative latest plant, these are outlined as follows:

Plant: 2F-2 (The date of commercial operation: February 3, 1984)

Operating cycle: 3rd cycle (May 24, 1986 - June 19, 1987)

Capacity factor during 3rd cycle: 98.6%

Number of MRF issued: average 1.2/day

Percentage of trouble classified by system:

Reactor system	15%
Turbine system	22%
Electric system	9%
Instrument system	53%

A percentage of MRF issuances in each system differs at each plant, however the common factor is that a percentage in the instrument system as well as the turbine system is relatively high compared with the other systems.

Therefore, we precisely analyze MRFs in the instrument system by classifying them into groups of a component, a phenomenon of troubles, a cause, a method of repair and a method of finding troubles, and also we precisely analyze them from a view point of transition in each operating cycle and comparison with the cases at other plants.

We are trying to prevent recurrence of problems and to insure the results of analysis provide timely feedback to the plant design and inspection standards (inspection cycle and method).

II-2 Maintenance management during periodical inspection

Japanese law specifies that the reactor system must be inspected every 12±1 months, and the turbine generator system must be inspected every 24±1 months, to maintain plant soundness.

Though the major modification work, which is represented by IGSCC countermeasure between 1975 and 1980, is almost completed, some modification work is still underway for some older plants in which the outage takes approximately 150 days.

The periodical inspection work for newly constructed plants is standardized resulting in an outage period between 80 and 90 days assuming no modification work is carried out. The electric utility begins to study the details of the work and schedule, with the plant manufacturer in charge of the major work in the outage, about 6 months before the start of periodical outage specified in the long term plant operation program.

The results of study are summarized in the form of a periodical outage work plan which requires approval by MITI about one month before the start of periodical outage.

(1) Types of work

Following types of work are carried out during a periodical outage in addition to refueling and fuel inspection.

a) In service inspection (ISI) of a reactor pressure vessel (RPV), piping of reactor coolant pressure boundary and so on.

b) Disassembly, inspection and repair of turbine, pumps, valves, heat exchangers, motors and so on.

c) Functional test of engineering safety features such as ECCS system, turbine protection mechanisms and safety relief valves.

d) Inspection, calibration, functional test of instrumentation in reactor protection systems and process systems.

e) Replacement of consumables such as valve gland packings, flange gaskets, pump mechanical seals.

f) Replacement of deteriorated equipment such as control rods and LPRMs.

g) Settlement of the trouble (MRF) that cannot be repaired during plant operation.

h) Deterioration monitoring by measuring thickness of feed water heater shell and pipes in the extraction system.

i) Modification work to improve plant reliability and to prevent recurrence of any problems in other plants.

j) Cleaning the plant and re-painting the floor.

(2) Volume of Work

Volume of the periodical outage work differs in each plant, but an approximate figure in Fukushima Daini Unit No.2 (2F-2), which is our representative plant of 1,100Mw, is as follows:

* Total man days	total	90,000 man/days
	peak	1,500 man/day
	average	1,000 man/day
* Number of tasks		1,200

* Major work items:

Disassembly, inspection, repair of pumps	100/360 sets
Disassembly, inspection, repair of valves	1,300/30,000 sets
Disassembly, inspection, repair of heat exchangers	50/140 sets
Disassembly, inspection, repair of motors	800/1,500 sets
Inspection and calibration of instrumentations	6,000/10,000 sets

(3) Inspection required by law

As for periodical inspections during the outage specified by the law, the standardized inspection procedure is established in which the periodical inspection execution procedure, stipulated by the government, and the concrete onsite inspection method for each type of reactor is described in detail based on past inspection experience.

It also describes the purpose of inspection, the extent of inspection, inspection items, inspection conditions, inspection methods and judgement standards.

The electric utility must prepare the periodical inspection procedure separately for each inspection work and must obtain MITI's approval.

In the case of BWR, the total number of inspection items is 71.

a) Witness items by MITI

* Items concerning the total plant performance.

* Items concerning the function of engineered safety features and function in reactor protection system.

* Other items considered important from a view point of safety.

Above items are subject to witness by MITI and the total number of witness items is 21.

The following items are subject to witness by a MITI inspector.

* Full load performance test

* Turbine performance test

* MSIV function test

* ECCS function test

* Function test of reactor protection system interlock

* Function test of stand by treatment system

* Leak test of primary containment vessel

* Isolation function test of primary containment vessel isolation valve

b) Record confirmation by MITI

Disassembly and inspection records of pumps and valves belonging to the engineered safety features and records of an emergency diesel generator are filed and checked by MITI.

However, these inspection items must be witnessed by Japan Power Engineering and Inspection Corp. (JAPEIC) which is to be the third party organization designated by MITI.

c) In case equipment is to be replaced or modification work is planned (see paragraph III), the electric utility must take the necessary procedure to revise the Construction Permit as required and pass the pre-use inspection of the hardware.

(4) Work organization

* In Japan, the original manufacturer who fabricated the critical equipment for securing safe and stable plant operation usually takes the responsibility for the maintenance of the equipment. Accordingly, the

main contractor of the plant construction (a full turn-key contract is popular in Japan) takes a leading part in the plant maintenance together with several other vendors.

* It's also the usual practice in Japan that the original equipment manufacturer takes the responsibility for spare parts supply, and thus the utility keeps only a few dozen specific spare parts in stock, which are imported from overseas manufacturer and have a long lead-time to procure.

* As the same maintenance work is usually undertaken by the same contractor in every outage, the maintenance work plan is established in accordance with the long term program and incorporates the experience and data accumulated by the utility and the contractor.

* The maintenance staff in the utility, which consists of 40 to 50 persons per 2 plants, is not directly involved in the actual field work, but involved in developing the work plan, investigating the contents of the work, work schedule control, witness of the work and evaluation of the work report.

* Presently, the outage work is conducted as a rule in a day time only (one shift from 8:00 AM to 17:00 PM) and the limited critical work is conducted based on two shifts (8:00 AM to 17:00 PM, 17:00 PM to 22:00 PM)

* When the outage starts, approximately 10 shift operators form a special group, called the outage group. In order to smoothly proceed with the outage work they coordinate outage schedule, equipment isolation and start/stop operation, safety procedure check of PTW (over 1500 items per outage), and make, install and remove more than 10,000 tags, with the close relation and contact with the maintenance staff.
(PTW: Permit to work)

(5) Countermeasure against the trouble at other plants

* Before the outage starts, the problems and failures which have occurred in Japan and overseas countries since the previous outage are carefully examined and checked if any inspection of similar equipment is required during the outage in order to prevent recurrence of similar problems. If we decide it is needed, additional inspection, modification and/or replacement is incorporated in the outage planning.

30 or 40 cases of trouble are usually selected for the investigation per outage.

* Before the outage starts, these work plans are required to obtain a MITI approval together with the outage work schedule, refueling plan and radiation control plan. And after completion of the outage, we report to MITI our evaluation for the deviation between the plan and actual, together with any trouble found during outage.

(6) Radiation control

* During the outage, several measures are taken to reduce radiation exposure to workers to as low as possible. (See paragraph IV)

* Before the outage starts, we establish the plan to reduce radiation exposure based on the maintenance work plan, such as installation of radiation shield, decontamination and use of automated equipment. We estimate the amount of radiation exposure before the outage and daily check the actual amount work by work during the outage. After completion of the outage, we compare the estimated amount with the actual amount and evaluate the difference. Any lessons will be reflected in the next outage.

* As a result of our effort described above, the total radiation exposure to workers during the outage has decreased and the level in the newest plant is 20 to 50 man-rem per outage.

(7) Modification work

* In addition to the inspection stipulated by law and the voluntary inspection based on the inspection standard made by utility, several kinds of modification work are performed during the outage. (See paragraph III)

These modifications are planned and investigated by the utility and the plant constructor jointly from a long range point of view, and these are classified according to priority for implementation.

* The modification work is classified and managed item by item, as follows, and a large part is voluntarily carried out by the utility.

Countermeasure for safety improvement

Countermeasure for reliability improvement

Countermeasure for radiation exposure reduction

Countermeasure for aged deterioration

II-3 Inspection standard

* In order to secure the reliable operation of the plant, the utility voluntarily specifies the inspection cycle and contents for each equipment, so called "Inspection standard" and carries out the inspections in accordance with such a standard. As a matter of course, the inspection stipulated by law is incorporated in the standard, and managed by uniform operation.

* "Inspection standard" is established taking into consideration the following factors: legal requirements, the degree of importance in terms of securing safe and stable plant operation, operation hours, service condition, system configuration, component life, operation and maintenance history, failure mode, condition of spare and replacement parts holdings, time required for inspection, accessibility, radiation exposure, and cost.

* The original inspection standard was established based on the recommendation by the plant manufacturer, our experience with fossil power plants, legal requirements and so on. Afterward, it has been revised from time to time to incorporate the operation and maintenance history and failure record of each equipment, taking into consideration the accumulation of our nuclear power plant operation experience over the past 10 years.

* The inspection standard consists of the following documents:

a. Inspection and overhaul standard.

Types of inspection and overhaul (Note 1), cycle of inspection (Note 2) and inspection contents are specified for each equipment.

b. Inspection cycle standard (10 years plan).

In accordance with the cycle of inspection defined in a. above, the control schedule is developed to specify the type of inspection in detail for each equipment in every outage for the next 10 years.

c. Inspection and overhaul standard classified by each system.

Contents of inspection for each equipment and component which constitutes the system are specified in detail.

d. Inspection and overhaul work procedure classified by each system.

Inspection and overhaul work procedure manual is prepared based on item c. above.

The maintenance contractor, who actually carries out outage works, prepares a more detailed work sequence in accordance with the above work procedure d. and obtains a work permit from the utility.

(Note 1) types of inspection and overhaul are classified into three categories and specify the degree of inspection.

Regular inspection...includes the inspection and overhaul of equipment, such as pumps and motors, overhaul of vessels, characteristic tests of instrumentation and control equipment and so on.

Simple inspection...includes the inspection of limited deteriorated component, such as motor bearing disassembly, pump shaft seals inspection, ground packing replacement of valves and so on.

Common inspection...includes insulation resistance tests of electric parts such as motors, oil supply, inspection of terminals and so on.

(Note 2) The cycle of inspection is determined depending on the types of equipment and the manner of inspection and overhaul. The inspection cycle of pumps and valves, which play an important role as mechanical equipment, is determined in accordance with the following view point.

1) Inspection cycle of pumps

The inspection cycle of pumps is determined taking into consideration the degree of importance and the frequency of operation, and whether pump is used for emergency use. Since there are many types of pumps for normal use, the cycle spreads over a wide range. The major pumps are inspected at intervals of a few years.

As pumps for emergency use have less accumulated operating hours than those for normal use, the inspection cycle is basically somewhat longer. The leakage check and activation test of each pump is carried out in every outage.

2) The inspection cycle of valves

Valves are classified into three categories, major valves which are described in Construction Permit, special valves, and other valves.

The major valves, which are mainly used for isolation, are disassembled and inspected approximately every 10 years depending on kinds of system, location, etc.

Among the major valves, the special valves such as SRV which are categorized as being essential are disassembled and inspected at shorter intervals of 1 to 4 years.

Other valves, which can be repaired during plant operation, are disassembled and inspected from time to time without specific intervals, and the inspection is carried out at the same time when related auxiliary equipment is inspected.

However, since the result of operation and maintenance for certain equipment is extremely satisfactory, recently we plan to revise and rationalize the inspection cycle, and exemplify the cases as follows:

MSIV disassembly and inspection
1 year - 4 years interval

MS.SRV disassembly and inspection
1 year - 2 years interval

CRD disassembly and inspection
5 years - 7-8 years interval

III. Measures to improve reliability and countermeasures against aged deterioration

(1) During the outage, various kinds of modification are carried out to improve the operability and reliability of the plant and to take countermeasures against aged deterioration.

In the planning stage of these modifications, any drastic permanent measures to be implemented are investigated with reference to the latest design introduced into our Fukushima Daini Unit 4 (2F-4) and our operation experience.

Major modification work which has been carried out recently is outlined below.

a. Low pressure turbine rotor replacement

b. Replacement of sea water piping which is buried directly outside a building.

c. Repair and painting of Mark I containment and modification of SRV discharge line.

d. Condensate pre-filter (hollow fiber filter) installation

e. Turbine EHC panel replacement

f. Process computer replacement

g. Feedwater control and recirculation control system replacement

h. Control rod position indication panel replacement

(2) While there are few problems leading to unexpected plant shutdowns, especially in the relatively older plants which have been operating for over ten years, some difficulties are found, especially in the control and instrument systems. One example of such trouble is that some replacement parts are difficult to procure. As most of these troubles show the aspect of aged deterioration, it is concluded that the control and instrument systems in the old plants may be close to the end of their useful lives.

Due to existing conditions, we have begun to introduce system renewal of the major control and instrument system considering problems that occurred in each plant during the past several years.

In the planning stage of system renewal, the latest plant design is modeled and triplexing of control system is adopted so as to improve reliability, operability, monitorability and serviceability.

A representative case of system renewal in the control and instrument system is Turbine EHC panel replacement.

The following countermeasures have been taken to improve the reliability of EHC panels.

* Multiplexing of sensors

* Multiplexing of control power supply

* Multiplexing of control circuit

In addition to the aforementioned countermeasures, an abnormal event diagnosis function for each printed wiring board, display on a monitor panel, reduction in the number of printed wiring boards by means of integration of electronic parts, and separation between trip circuits and computing circuits help to improve monitorability, operability and maintenanceability.

IV. Several measures to secure quality during maintenance

(1) In order to secure reliability by means of preventive maintenance, it is required to assure that the quality level of the maintenance work itself shall be high, and that the possibility of unnecessary problems due to an error during maintenance work be minimized.

In Japan, various kinds of effort have been made to secure high quality during maintenance work and improve reliability.

The major items are outlined below.

a. Improvement in the access to working area and the working environment is made by plant design change, such as the space extention in the primary containment vessel and the establishment of private room exclusively used for SRV inspection and overhaul.

b. Improvement in the working environment is made by several measures to reduce radiation exposures, such as the completeness of water chemistry control, material change, and radiation shield reinforcement.

c. Introduction of remote automated equipment, such as the automated refueling machine and automated CRD exchanging machine.

d. Improvement of education and training for the maintenance staff to help improve their capability, such as the establishment of a maintenance training center equipped with several full-scale mockups.

e. Improvement of the manuals used for maintenance work.

f. Establishment of the system where the plant manufacturer consistently takes the responsibility for quality from construction thru operation of the plant.

(2) These measures are the result of a joint research and development effort performed by the utility and the plant manufacturer using a long range point of view. Part of them was investigated in the improvement and standardization program which has proceeded from 1975 under the direction of our regulatory authority, and some of them were implemented in the program.

V. Future tasks of maintenance management

Today, the maintenance management of nuclear power plant in Japan is extremely good, and we consider that such excellent management contributes to the high capacity factor and low scram frequency in recent years. However, as represented by a rather long outage period of three months, it does not always follow that we carry out the maintenance management rationally.

In the future, while we maintain the existing high reliability, we need to carry out the necessary maintenance systematically and in the most appropriate time so as to promote more effective equipment operation with the intention of plant life extension.

Based on the above, we will intensively promote the following program from this point on.

a. In order to secure quality in maintenance, we will improve man-man and man-machine interfaces in maintenance work as well as capability of the maintenance staff.

b. We will positively promote to develop deterioration diagnostic technology, inspection and repair technology, etc. since countermeasures for aged deterioration need to be timely implemented with the increasing years of plant operation.

c. In order to establish the rational inspection standard which properly reflects the operation and maintenance experience, we need to develop a data base based on our accumulated operation and maintenance data.

LIFE EXTENSION LESSONS FROM BWR OPERATION

R. J. Brandon, G. M. Gordon, E. Kiss, S. Ranganath, P. P. Stancavage

GE - Nuclear Energy

Abstract

Field experience with operating boiling water reactors (BWRs) over the past twenty-five years constitutes a wealth of information useful for plant life extension. This operating data demonstrates large margins in many BWR components in support of operation beyond the nominal design life, for example, reactor pressure vessel toughness after irradiation. For some components, experience has sparked the development of technology to counter problems such as stress corrosion cracking in which new materials and water chemistry control serve to mitigate age related degradation.

This paper summarizes BWR operation from the perspective of life extension. Methods for identifying, measuring, mitigating and correcting the effects of life limiting mechanisms are described. The focus is on proven effectiveness for the BWR pressure vessel and containment.

Identify Issues

Identifying key issues for life extension means knowing where to look. This includes the ASME Boiler and Pressure Vessel Code, Regulatory Guides, design stress reports and Safety Analysis Reports. All these documents reveal which items were considered in the original design of the plant. In some cases, enough information is available to determine how close acceptance criteria were approached. However the design basis documentation does not tell the whole story because, for example, fatigue calculations are often only refined until the criterion of a usage factor of 1.0 is met.

Operating experience enriches this codified body of knowledge. Most observations confirm our expectations but there have been occasional surprises. We need to look at other plant experiences and even other industries. So long as we have a firm grasp of the underlying physical phenomena and a knowledge of the operating environment, our identification of the key issues is on a sound footing. Consider, for example, feedwater nozzle leakage and stress corrosion cracking in creviced Inconel 182. In both these cases, field incidents called attention to the phenomena and motivated laboratory experiments along with fundamental research which led to effective remedies.[1,2] A comprehensive data base is probably the single most important element of correct identification of the mechanisms and components because it allows systematic processes such as root cause analyses to proceed with a broad foundation of experience.

Table 1 shows a matrix of the major mechanisms affecting BWR longevity based on life extension studies[3,4] tempered with operating experience.

Table 1

BWR Age Related Degradation Priority

	Phenomenon			
Component	Stress Corrosion Cracking	Neutron Embrittlement	Fatigue	Corrosion
Pressure vessel	Moderate	Low	High	Low
Vessel internals	High	Low	Moderate	Low
Containment	Low	Low	Moderate	High

Fatigue is most important for the reactor pressure vessel due to the continuous injection of relatively cool feedwater during power operation and the intermittent injection of cold water during shutdown. There is a moderate concern for stress corrosion cracking in the vessel, primarily due to the presence of safe ends and attachment welds which are susceptible to stress corrosion cracking. The low alloy steel of the BWR shell resists stress corrosion cracking. Surveillance data and corrosion tests suggest that the loss of fracture toughness after neutron exposure and the oxidation rate are relatively minor contributors to determining the overall service life of the vessel.

Vessel internals, primarily constructed of 304 stainless steel, have shown incidents of stress corrosion cracking particularly where crevices exist, weld residual stresses are high, neutron flux is large and susceptible material is used. There are some indications that fatigue has caused degradation of components, for example, jet pump instrument lines, but for most internal components loads are well below the endurance limit. Neutron embrittlement does not affect internals material and, based on experiments with irradiated reactor components, fracture toughness remains high for most internals outside of the fuel region. Corrosion is minimal due to the use of stainless steel.

The containment, fabricated in many plants of carbon steel with no corrosion allowance, has shown itself to be most susceptible to the traditional phenomena of corrosion and fatigue. Corrosion appears in the pressure suppression chamber when the coating has deteriorated and in the drywell where moisture can accumulate. Fatigue has resulted in some signs of wear near pipe attachment points due to vibration.

Field experience with operating BWR plants results in the identification of the key mechanisms and susceptible locations which are essential to developing a credible life extension program.

Measure Condition

Condition measurement includes both off-line techniques such as inspection which are the basis for most of the current practices in the U.S. and on-line techniques such as fatigue and stress corrosion monitoring[5] which are being more widely adopted. Measurement can also provide immediate benefits, for example, mid-cycle shutdown avoidance. The measurements should be integrated so an overall picture of the status of a component can be drawn. Process variables are effectively used as an overall measure of degradation because material deterioration often causes anomalous performance.

Successful examples of measurement include the ultrasonic testing of wall thickness for the carbon steel containment in the vicinity of the drywell floor, fatigue monitoring of the vessel feedwater nozzle, surveillance specimen testing to determine fracture toughness, stress corrosion monitors to determine crack growth and water chemistry monitoring to determine conformance with appropriate guidelines.

Condition assessment of key components provides the means to confirm structural margin during normal and extended service life. It also forms the basis for making long term decisions affecting the longevity of the nuclear power plant.

Mitigate Degradation

Mitigating age related degradation includes those activities taken before degradation proceeds far enough to warrant repair or replacement. For stress corrosion cracking, there is water chemistry control including hydrogen addition, stress reduction by, for example, post weld heat treatment or induction heating, and material improvement such as using 316NG stainless steel[6]. For fatigue there is leakage prevention by improved thermal sleeve design, clamping parts to reduce mechanical stress, reducing the frequency of plant transients and improved feedwater control systems which are effective at low flow rates. Corrosion has been effectively mitigated by coating systems and by cathodic protection in areas where painting is not appropriate. Low leakage cores have been designed for PWRs and can be applied to BWR plants if the need arises.

Mitigation can substantially lengthen the life of a BWR plant by slowing or preventing age related deterioration for important components.

Refurbish Equipment

Correcting deterioration by repair or replacement is the final step in assuring the life of components when sufficient margin cannot be demonstrated for the extended service life. Several examples of repairs include clamps on core spray spargers, weld overlays for recirculation system piping, and mechanical seals on control rod drive penetrations. Replacements include shroud head bolts, jet pump restraints, stainless steel pipe and safe ends and quenchers installed in place of rams head devices in the containment.

Corrective action must focus on durability, not just on a quick repair and return to power. Durability means more than just a higher grade of materials, it also implies resistance to all known or suspected mechanisms, a long history of application in a similar environment and precise work during installation.

Equipment refurbishment restores components to the beginning of their lives, and, if durable, ensures a longer life expectancy than the original equipment.

Summary

BWR operating experience over the past twenty-five years enriches the knowledge documented in codes, regulations and safety analyses so that life extension can be pursued with confidence. Research stimulated by field incidents has promoted understanding of the key mechanisms affecting the longevity of the reactor vessel, internals and containment. Monitoring, inspection and surveillance are continually improving our assessment of the condition of major BWR structures and equipment. Several techniques have been developed to mitigate age related degradation due to stress corrosion cracking, corrosion and fatigue. Experience has proven valuable in demonstrating the feasibility and worth of durable repairs and replacements.

References

1. U.S. NRC, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," NUREG-0619.
2. F. P. Ford, et al., "Corrosion Assisted Cracking of Stainless and Low Alloy Steels in LWR Environments," EPRI NP-50645, February 1987.
3. "BWR Pilot Plant Life Extension Study at the Monticello Plant: Phase 1, Northern States Power," EPRI NP-5181M, May 1987.
4. R. J. Brandon, P. P. Stancavage, "For BWR longevity, plants are the best teachers," Proceedings of an international symposium on safety aspects of the ageing and maintenance of nuclear power plants, IAEA, Vienna, Austria, July 1987.
5. E. Kiss, S. Ranganath, "On-line monitoring to assure structural integrity of nuclear reactor components," Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, Post SMIRT Conf. Brussels, Belgium, 1987.
6. B. M. Gordon, G. M. Gordon, "Material Aspects of BWR plant life extension," Nucl. Eng. Des. 98, pp. 109-121, 1987.

CHEMICAL CLEANING OF STEAM GENERATORS AT HIGH TEMPERATURES

by Dr. Klaus Froehlich and W. R. Greenaway

Introduction

Steam generator degradation is one of many factors that must be considered when evaluating power plant aging. Three plants in the United States have already had their steam generators replaced due to unexpected corrosion damage,and at least three more are under serious consideration for the same action. The damage to the steam generators in all cases has been the result of the ingress of impurities during operation and their concentration, by boiling, to very corrosive levels in areas of poor circulation. Contributing to this concentration mechanism are large quantities of metal oxides resulting from the surface corrosion of steam, condensate, and feedwater system materials and the sloughing off of these oxides into the feedwater. Sometimes these oxides are a prerequisite to the corrosion process.

The oxides entering the steam generator accumulate on the tube surface, on the horizonal support plate surfaces, in tube-tube support crevices, and in a large pile on the tube sheet. Normally, only the tube sheet sludge pile is routinely removed, by high pressure water lancing. Minor amounts of sludge are sometimes also removed during certain crevice flushing operations designed primarily to remove soluble salts. But the largest amount of sludge is actually on the tube surface because of the large surface area. Sludge flakes taken from the tube sheet during lancing often indicate under examination that they originated on the tube surfaces and were dislodged by some thermal or mechanical shock. Thus, removing sludge from the tube sheet really attacks only a very small portion of the overall steam generator sludge accumulation; sludge flakes will continue to contribute to the tube sheet sludge pile buildup during the next cycle of operation. Only total sludge removal by chemical cleaning will effectively keep the steam generator clean for at least several cycles and prevent tube surface attack from impurities concentrating in the sludge layers. Careful attention to steam, condensate, and feedwater system chemistry is, of course, the key to lengthening the periods between sludge removal operations.

Unfortunately for some plant owners, the tube-denting damage mechanism has progressed to a point where it is no longer feasible to chemically clean above the first support plate. The oxides grown on the support plate surfaces are the only "glue" holding the support plates together. Removal of this material may result in loose pieces of metal circulating in the steam generators during operation, a highly undesirable condition for tube integrity. Utilities with this condition, unfortunately, are left with only steam generator replacement as an alternative, the time for which may be extended by such temporary measures as boric acid feed and/or much greater emphasis on steam generator operating chemistry.

For those utilities whose steam generators are not severely degraded, chemical cleaning processes can effectively remove the sludge accumulation. The best known of these is the one developed through EPRI funding and commonly referred to as the SGOG process. This two-phase process is applied at a temperature below the atmospheric boiling point and requires extensive external process equipment to apply. So far, the SGOG process has been used to remove sludge piles from the tube sheets of two steam generators and for complete sludge removal in four others. The time of application varies with the number of process steps performed, but has taken as long as 1 week. Waste treatment has been a problem in most cases.

The KWU Process

At about the same time that the EPRI-funded program began, KWU began evaluating several candidate processes in Europe. This work consisted of solvent optimization studies on numerous sludge compositions using many different chemical approaches. Lab-scale tests were run, and large scale model boilers were used in the program. The process developed was adequate, but still had undesirable properties. In late 1985, KWU became aware of a process routinely used by the Institue für Energetik, Leipzig, for cleaning fossil boilers and that had been used in their PWR program. KWU reviewed the applications of this process and set up a testing program to evaluate it with respect to cleaning a KWU-designed U-tube steam generator. The results of this test series were so encouraging that work on the original KWU process was halted and the new process was adopted.

A full-scale qualification program followed. This culminated in the cleaning of one of the U-tube steam generators at the Neckarwestheim plant in the summer of 1986. Since then, a plant-specific qualification program was performed for the Almaraz Plant (Westinghouse design), and in December 1986 the three steam generators at this plant were cleaned. Numerous applications in Germany, Spain, and Sweden have followed since.

As with the EPRI process, two distinct phases are involved in the KWU sludge removal process. In the first, the copper is removed; in the second, the iron. If it is more convenient, these processes can be reversed.

Copper Removal Phase

The copper solvent is an ammonia-based material that performs its dissolution function at low temperature and high pH (> 9.5). The application temperature is less than 140°F and is easily controlled, since no exothermic reactions are involved. Oxidation of metallic copper in the sludge is accomplished by air in the presence of a catalyst that allows sufficient oxidation without the use of the more common oxidizers, such as hydrogen peroxide. The solvent contains no chelants, although a compound similar to the familiar ammonia/copper complex is formed and remains stable until broken in the waste treatment process.

The concentration of the solvent application depends on the amount of copper/copper oxide to be dissolved. Generally, the concentration is in the 2-to-4 percent range. Because the chemical constituents of the copper solvent are ammonia based, they decompose when the plant heats up and can be boiled off. Any residual is the original copper, which is released when the complex is broken at the high temperature.

Copper Solvent Corrosion

With the low application temperature and high pH, there is little corrosion of steam generator materials. However, to confirm this, testing was done at a higher temperature (160°F) with a solvent contact times of 24 hours and longer.

Tables 1 through 3 show the results of these tests on a variety of materials and material combinations The results indicate corrosion of less than 0.2 mils for all materials and combinations tested except for one specimen of SA 564 with hardfacing. Even in this case, the attack at the hardfacing interface was only about 1 mil. There were no indications of significant galvanic effects or HAZ attack on other materials. In the galvanic couples it was observed that the anodic material did have a slightly higher weight loss in some cases but was still under the 0.2-mil total corrosion.

TABLE 1. CORROSION TEST RESULTS FOR COPPER REMOVAL: CARBON STEEL AND LOW ALLOYED MATERIALS

German Standards	ASME	Localized Corrosion (mils per event)	General Corrosion (mils per event)
St 37	SA 442 Gr. 55 (plates)	< 0.2	< 0.2
15 Mo 3	SA 182 F 1 (forgings)	< 0.2	< 0.2
20 MnMoNi 55	SA 508 Cl. 3 (forgings)	< 0.2	< 0.2
22 NiMoCr 37	SA 508 Cl. 2 (forgings)	< 0.2	< 0.2
GS-17 CrMoV 511	SA 217 WC 11 (casting)	< 0.2	< 0.2
17.4 PH	SA 564 Type 630	< 0.2	< 0.2
X 5 CrNi 13 4 (1.4313)	SA 564 Type 630	< 0.2	< 0.2
X 20 CrNi 17 2 (1.4057)	SA 182 F 6	< 0.2	< 0.2
X 35 CrMo 17 (1.4122)	ASTM A 276 Type 440 A	< 0.2	< 0.2
X 10 Cr 13 (1.4006)	ASTM A 276 Type 410	< 0.2	< 0.2
X 90 CrMoV 18 (1.4112)	ASTM A 276 Type 4400	< 0.2	< 0.2

TABLE 2. CORROSION TEST RESULTS FOR COPPER REMOVAL: HIGH ALLOYED STEELS

German Standards	ASME	Localized Corrosion (mils per event)	General Corrosion (mils per event)
SKWAM (17% Cr)	SFA-5.4 E 430	< 0.2	< 0.2
Stellite 6	SFA-5.13 E CoCr-A	< 0.2	< 0.2
Combination Stellite 6 - 20 MnMoNi 55	SFA-5.13 E CoCr-A/ SA 508 Cl. 3	< 0.2	< 0.2
X 5 CrNi 18 9 (1.4301)	SA 240 Type 304	< 0.2	< 0.2
G-X 6 CrNi 18 9 (1.4308)	SA 351 CF 8 C	< 0.2	< 0.2
G-X 6 CrNiMo 18 10 (1.4408)	SA 351 CF 3 M	< 0.2	< 0.2
INCONEL 600	SB 163 NiCrFe	< 0.2	< 0.2
INCONEL 182	Comparable, Inc. 600	< 0.2	< 0.2
INCONEL 718	ASTM-B 670	< 0.2	< 0.2
INCOLOY 800	SB 163 NiFeCr	< 0.2	< 0.2

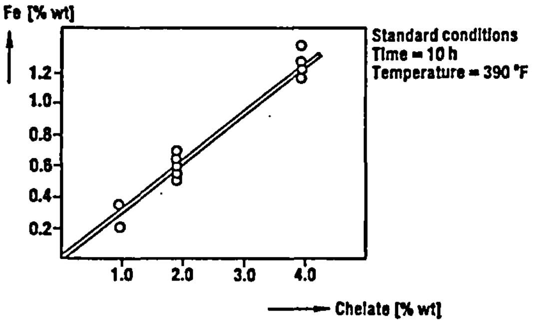
TABLE 3. CORROSION TEST RESULTS FOR COPPER REMOVAL: COMBINATIONS AND WELDINGS

German Standards	ASME	Localized Corrosion (mils per event)	General Corrosion (mils per event)
St 37	SA 442 Gr. 55 (plates)	< 0.2	< 0.2 welding
15 Mo 3	SA 182 F 1 (forgings)	< 0.2	< 0.2 welding
20 MnMoNi 55	SA 508 Cl. 3 (forgings)	< 0.2	< 0.2 hardfaced with Stellite 6
22 NiMoCr 37	SA 508 Cl. 2 (forgings)	< 0.2	< 0.2 welding
X 5 CrNi 134 (1.4313)	SA 564 Type 630	≈ 1	< 0.2 hardfaced with SFA-5.4 E 430
SKWAM (17% Cr)	SFA-5.4 E 430	< 0.2	< 0.2 hardfaced with SA 564 Type 630
Stellite 6	SFA-5.13 E CoCr-A	< 0.2	< 0.2 with SA 508

Iron Removal Phase

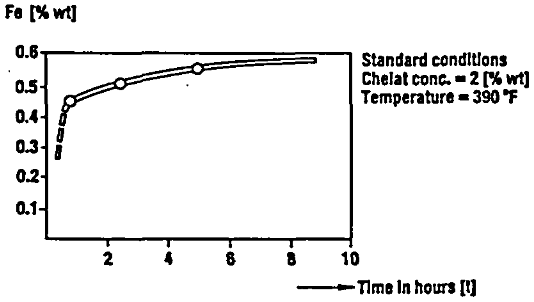
The iron solvent is composed of a nonvolatile chelant, a pH controlling additive, and a reducing agent. The application temperature has been in the 300-to-400°F range, although it will work, perhaps somewhat more slowly, at lower temperatures. Figure 1 provides a graph of the dissolution properties of the solvent for iron oxides relative to its concentration. Typical parameters for application are between 2 and 4 percent (20 to 40 g/l), but can be significantly higher. As with the copper process, the actual concentration will be determined by the amount of iron oxides to be removed. Figure 2 demonstrates the dissolution kinetics of the process and shows that the majority of the dissolution occurs in the first hour. Figure 3 shows the relationship between the application temperature and the dissolution rate. This curve is influenced by two factors. In the region from 210 to 390°F, the dissolution rate, and thus the maximum solvent loading during the test time, is increased due to the faster reaction kinetics resulting from increased temperature. Above 390°F, the effectiveness drops off due to decomposition of the solvent.

This solvent decomposition is significant, because any trace amounts remaining in the system after the cleaning application will rapidly decompose to acetic acid, which will in turn decompose to carbon dioxide and water, resulting in no long-term residuals.



Experience from various applications
$\frac{\text{Chelate conc. (\% wt)}}{\text{Iron conc. (\% wt)}} = 3.4$

FIGURE 1. IRON DISSOLUTION AS A FUNCTION OF CHELATE CONCENTRATION



Conclusion
Application time depends only on the time required to treat the solvent in the SG:

- Chemical feed
- Heat-up
- Mixing
- Cool down
- Draining etc.

FIGURE 2. IRON DISSOLUTION AS A FUNCTION OF TIME

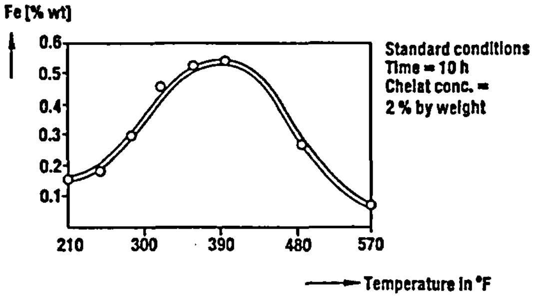


FIGURE 3. IRON DISSOLUTION AS A FUNCTION OF TEMPERATURE

Iron Solvent Corrosion

As would be expected, the corrosion from the iron dissolution step is somewhat greater than from the copper removal step. The qualifications tests were performed with a chelant concentration of 2-to-4 percent and temperatures of 340 to 390°F. Coupons without a protective layer of iron oxides were used to establish the most severe corrosion conditions (the chelant loaded with iron is noncorrosive). To establish conservative contact times, all tests were performed with a 3-hour heat-up, 10 hours at temperature, and a 10-hour cooldown time. This is in excess of the time needed for sludge dissolution (2 hours) but provides a realistic time to get the solvent in and out of the steam generator. The results on a variety of materials and materials combinations are shown in Tables 4 through 6. As with the copper solvent, little localized corrosion was found. General corrosion was higher, reaching a maximum of 7 mils on one material combination. This combination does not exist in the steam generator, but may be present in valves on the secondary side.

TABLE 4. CORROSION TEST RESULTS FOR IRON REMOVAL: CARBON STEEL AND LOW ALLOYED MATERIALS

German Standards	ASME	Localized Corrosion (mils per event)	General Corrosion (mils per event)
St 37	SA 442 Gr. 55 (plates)	< 0.2	1
15 Mo 3	SA 182 F 1 (forgings)	< 0.2	1.3
20 MnMoNi 55	SA 508 Cl. 3 (forgings)	< 0.2	3
22 NiMoCr 37	SA 508 Cl. 2 (forgings)	< 0.2	1.5
GS-17 CrMoV 511	SA 217 WC 11 (casting)	< 0.2	1.3
17.4 PH	SA 564 Type 630	< 0.2	< 0.2
X 5 CrNi 13 4 (1.4313)	SA 564 Type 630	1	< 0.2
X 20 CrNi 17 2 (1.4057)	SA 182 F 6	< 0.2	0.5
X 35 CrMo 17 (1.4122)	ASTM A 276 Type 440 A	< 0.2	< 0.2
X 10 Cr 13 (1.4006)	ASTM A 276 Type 410	< 0.2	< 0.2
X 90 CrMo V 18 (1.4112)	ASTM A 276 Type 4400	< 0.2	< 0.2

TABLE 5. CORROSION TEST RESULTS FOR IRON REMOVAL: HIGH ALLOYED MATERIALS

German Standards	ASME	Localized Corrosion (mils per event)	General Corrosion (mils per event)
SKWAM (17% Cr)	SFA-5.4 E 430	< 0.2	< 0.2
Stellite 6	SFA-5.13 E CoCr-A	< 0.2	< 0.2 (some tests, ≈ 0.2)
Combination Stellite 6 - 20 MnMoNi 55	SFA-5.13 E CoCr-A/ SA 508 Cl. 3	< 0.2 on Stellite	< 0.2 on Stellite
X 5 CrNi 189 (1.4301)	SA 240 Type 304	< 0.2	< 0.2
G-X 6 CrNi 189 (1.4308)	SA 351 CF 80	< 0.2	< 0.2
G-X 6 CrNiMo 1810 (1.4408)	SA 351 CF 3 M	< 0.2	< 0.2
INCONEL 600	SB 163 NiCrFe	< 0.2	< 0.2
INCONEL 182	Comparable, Inc. 600	< 0.2	< 0.2
INCONEL 718	ASTM-B 670	< 0.2	< 0.2 (some tests, 0.3)
INCOLOY 800	SB 163 NiFeCr	< 0.2	< 0.2

TABLE 6. CORROSION TEST RESULTS FOR IRON REMOVAL: COMBINATIONS AND WELDINGS

German Standards	ASME	Localized Corrosion (mils per event)	General Corrosion (mils per event)
St 37	SA 442 Gr. 55 (plates)	< 0.2	≈ 3 welding
15 Mo 3	SA 182 F 1 (forgings)	< 0.2	≈ 3 welding
20 MnMoNi 55	SA 508 Cl. 3 (forgings)	< 0.2	≈ 7 hardfaced with Stellite 6*
22 NiMoCr 37	SA 508 Cl. 2 (forgings)	< 0.2	≈ 3 welding
X 5 CrNi 134 (1.4313)	SA 564 Type 630	≈ 1	≈ 4 hardfaced with SFA-5.4 E 430*
SKWAM (17% Cr)	SFA-5.4 E 430	< 0.2	≈ 3 hardfaced with SA 564 Type 630*
Stellite 6	SFA-5.13 E CoCr-A	< 0.2	≈ 0.4 with SA 508

*BOP, not steam generator combination

Note, however, that on most carbon steels and low alloy materials the corrosion rate is about 3 mils or less. Although not shown in the tables, other testing has shown that galvanic effects are detectable under some conditions but only cause an increase in general corrosion up to a factor of about 1.5 times.

Crevice Cleaning Studies

Because KWU's steam generator design does not experience denting phenomena at the tube support plates, extensive crevice corrosion studies have not been performed. However, this issue was investigated as part of the generic and plant-specific work done. As expected, the normal solvent application times are not sufficient to fully clean crevices. For an approximately 1-inch crevice around a tube, the corrosion products were partially removed during the iron solvent test application. It is noteworthy that even though iron oxides were remaining, subsequent analysis indicated that the entrapped salts had been essentially completely removed.

Plant-Specific Qualification Program

Recommended Solvent Optimization Procedures

Due to the extensive testing done to date and the successful application of the process, no extensive R&D effort is normally required for recirculating steam generators prior to use of the solvent at a given plant. What is recommended, however, is a short series of plant-specific qualification tests to optimize the solvent for the plant-specific corrosion products and to perform a demonstration of the exact process to be applied at a given plant on a model large enough to accurately simulate the corrosion potentials existing in the actual steam generator. Included in each of these test series would be base material and weldment coupons to verify that the corrosion to be expected is within acceptable limits.

The solvent development program consists of two major efforts: optimization of the dissolution properties of the solvent and materials effects testing relative to specific materials of construction for the PWR steam generator and associated feed train components.

Optimization studies are carried out using actual plant sludges. The amount removed by the solvent depends on the solvent capacity, which is determined by the concentration of the chelating agent, the pH-value of the solvent, the time of application, and the application temperature. It is also noteworthy that these are the same parameters that influence the general corrosion rate of the low and unalloyed carbon steels typically used in steam generator construction.

Typical Plant-Specific Test Program

A typical plant-specific program would consist of the following basic tasks:

1. Compare plant materials with existing database of previously tested corrosion specimens.
2. Review the corrosion product characterization.
3. Perform bench scale tests for solvent optimization.
4. Have an independent party review the program.
5. Develop the plant-specific chemical application specification.
6. Develop the chemical procurement specification.
7. Develop conceptual engineering flow paths, drawings, and procedures.
8. Perform full process demonstration test.
9. Perform waste qualification test.
10. Prepare final qualification report.

Each of these tasks is discussed in turn in the following paragraphs.

Compare Plant Materials with an Existing Database of Previously Tested Corrosion Specimens. Plant materials-of-construction data are compiled by the utility or the steam generator manufacturer for use as the basis for the materials review for this task. These materials are compared with those previously tested for corrosion in generic and plant-specific testing. A list of plant materials, materials tested, test results, and materials that require further testing is formulated during this task.

Review the Corrosion Product Characterization. Corrosion product samples from the plant are analyzed for composition. This information is compared with the sludge characteristics of plants previously cleaned and the corrosion product characteristics used in the solvent test programs. The final demonstration tests are based on the results of this review.

Perform Bench Scale Tests for Solvent Optimization. The bench scale tests are performed with actual plant corrosion products and consist of tests of the iron and copper solvents. These tests are run in autoclaves that have provisions for on-line sampling and chemical additions. The major thrust of these test series is to determine the solvent concentrations best suited for the plant-specific corrosion products and to verify solvent performance on the actual plant sludge. Included with these tests are the initial general corrosion studies on those materials requiring further investigation. These tests include incorporation of selected material test coupons into the autoclaves; post-cleaning, weight change evaluation for general corrosion; and microscopic examination for localized attack.

Have an Independent Party Review the Program. To provide assurance that complete qualification of this solvent system is obtained, the utility may wish to retain an independent consultant to review the material test results.

Develop the Plant-Specific Chemical Application Specification. An application specification is formulated based on all previous test results. This specification is the basis for developing the plant cleaning documents and performing the full process demonstration test. This document has complete information relative to solvent composition, application sequence, sampling and analysis required, corrective action for off-normal conditions, process termination criteria, chemical mixing criteria, and other pertinent information.

Develop the Chemical Procurement Specification. A chemical procurement specification is formulated based on the chemicals used in the test program and corresponding to those used to develop the generic database. This specification is used to purchase the chemicals for the demonstration test program and for the actual plant application.

Develop Conceptual Engineering Flow Paths, Drawings, and Procedures. This task consists of a complete review of the site equipment that would be used for the cleaning and a determination of the plant operational restrictions on the process. Basic flow diagrams for the systems, both installed and portable, are developed. A chemical application process guideline document is formulated based on the chemical application specification and the actual plant conditions. This document is later used to develop the full demonstration test procedure. The application guideline includes a description of the plant equipment to be used, the portable equipment to be used, and the support functions such as chemical storage, waste hold-up, and waste processing.

Perform Full Process Demonstration Test. After completing the other phases of the program, a full-process demonstration test could be performed in a large-scale SG model. This basic approach was developed under other qualification programs previously performed, but is sometimes omitted. The major purpose of the test is to test the actual plant solvent chemistry and procedures on a model large enough to accurately simulate the conditions in the steam generator. A second significant issue is to test the effects or lack of effects of galvanic corrosion on selected weldment specimens directly coupled to the model vessel.

Perform Waste Qualification Test. Two basic approaches can be taken in the area of waste handling. The first is to concentrate the waste, with direct solidification of the resultant concentrate. The second is to precipitate the metals from the solvent, followed by purification of the resultant liquid and drying or solidification of the precipitated materials. This task includes:

- Demonstration of the precipitation process, the purification of the resultant liquids, and stabilization of the sludge on a small scale
- Demonstration of the concentration process and stabilization of the resultant concentrate on a small scale
- Evaluation of available equipment to perform either the precipitation or the concentration process
- Performance of a cost analysis for each process, including processing, transportation, and disposal costs

These techniques have been developed to reduce both the chemical constituents and the radiochemical contaminants to concentrations acceptable for disposal under federal EPA and NRC requirements. Regional and state requirements may, in some cases, be more restrictive, and additional treatment may be required.

Prepare a Final Qualification Report. A final report is issued containing the data developed during the test programs and recommendations for the actual plant application.

Application Technique

The solvent process is designed to make maximum use of installed plant equipment. Generally, the only portable equipment required would be tanks for waste holding and treatment.

The cleaning sequence strongly depends on the copper content of the sludge in the steam generators. If the copper content exceeds about 10 to 15 percent, a copper removal step is usually performed first. If the copper content is less than 10 to 15 percent, the iron removal step should come first.

The process normally consists of an iron oxide cleaning step applied to the steam generators immediately following plant shutdown, but before cooldown, and a copper cleaning step applied in the cold shutdown condition. However, if copper-rich deposits in the steam generators make it preferable to begin with the copper step, then the system must be heated from ambient temperature to about 350°F before subsequent application of the iron step. Therefore, it may be desirable to perform the iron removal step as the plant is brought back on line after the shutdown. During the process qualification program, the exact solvent concentrations and the exact number and sequence of solvent application steps will already have been determined.

Iron Solvent Application

Concentrated chemicals are mixed in a portable mix tank or in a designated plant tank (e.g., a steam generator chemical dosing tank) prior to plant shutdown or prior to heating up to 300 to 400°F. The concentration is such that the final mixture in the steam generators is approximately 2 to 4 percent, depending on the results of the solvent optimization task of the process qualification program. Samples are taken, and the composition is verified.

After plant shutdown, the steam generators are cooled to approximately 400°F, or the upper limit for use of the residual heat removal system (RHRS). The iron solvent is then injected into the steam generators using an installed chemical injection pump or a portable high pressure injection system. This injection can be performed over a period of up to 2 hours.

The injection point is determined during the plant conceptual design study, but is anticipated to be either through the auxiliary feed header or backward through the normal steam generator drain or blowdown path.

The chemicals are allowed to soak in the steam generator for about 2 to 4 hours. During this time there are periodic boil-offs through the steam bypass system to provide for agitation of the solvent. During these periodic boil-offs no chelants are lost, due to their nonvolatile character. When the samples indicate that the solvent is depleted or the rate of iron oxide dissolution is approximately constant, the steam generator cooldown can commence.

After the cleaning process, the solvent is drained from the steam generator and a rinse solution applied.

The use of the RHRS to maintain the temperature in the steam generators during the cleaning process allows the application of the iron step to all steam generators in parallel.

Copper Solvent Application

The copper solvent can be applied at any time during the cold shutdown condition. However, since it has been shown that copper within the deposits on the steam generator tubes has a negative impact on the eddy current signals and their interpretation, the copper cleaning is recommended before the inspection of the steam generator. Prior to solvent application, an air sparger system is sometimes installed through the lower handholes. An additional option is to sparge through the blowdown line. The steam generator is then filled to the required level. The premixed concentrated solvent is injected into the steam generator, and the air sparging started. Periodic sampling is performed to monitor the cleaning progress. The entire process takes approximately 24 hours.

At the completion of the cleaning process, the copper solvent is drained from the steam generators to the waste holding system. (In Germany, it can remain in the steam generators as the wet lay-up chemistry.)

After the spent solvent is drained from the steam generator, a single volume rinse is performed.

Leaking Steam Generator Tubes During Chemical Cleaning

During the iron removal step, a leak from secondary to primary side cannot occur, because the primary system is at high temperature and pressure. However, if, during the copper removal process, a tube leak should occur, the cleaning solvent could enter the primary side. This is of no consequence to the primary side materials, because the solvent is not corrosive, is volatile, and thermally degrades as soon as the temperature is raised. The only possible concern is the introduction of a small amount of copper into the primary system. If the primary system can be kept at a slightly higher pressure, even this possibility can be avoided.

Cleaning Results

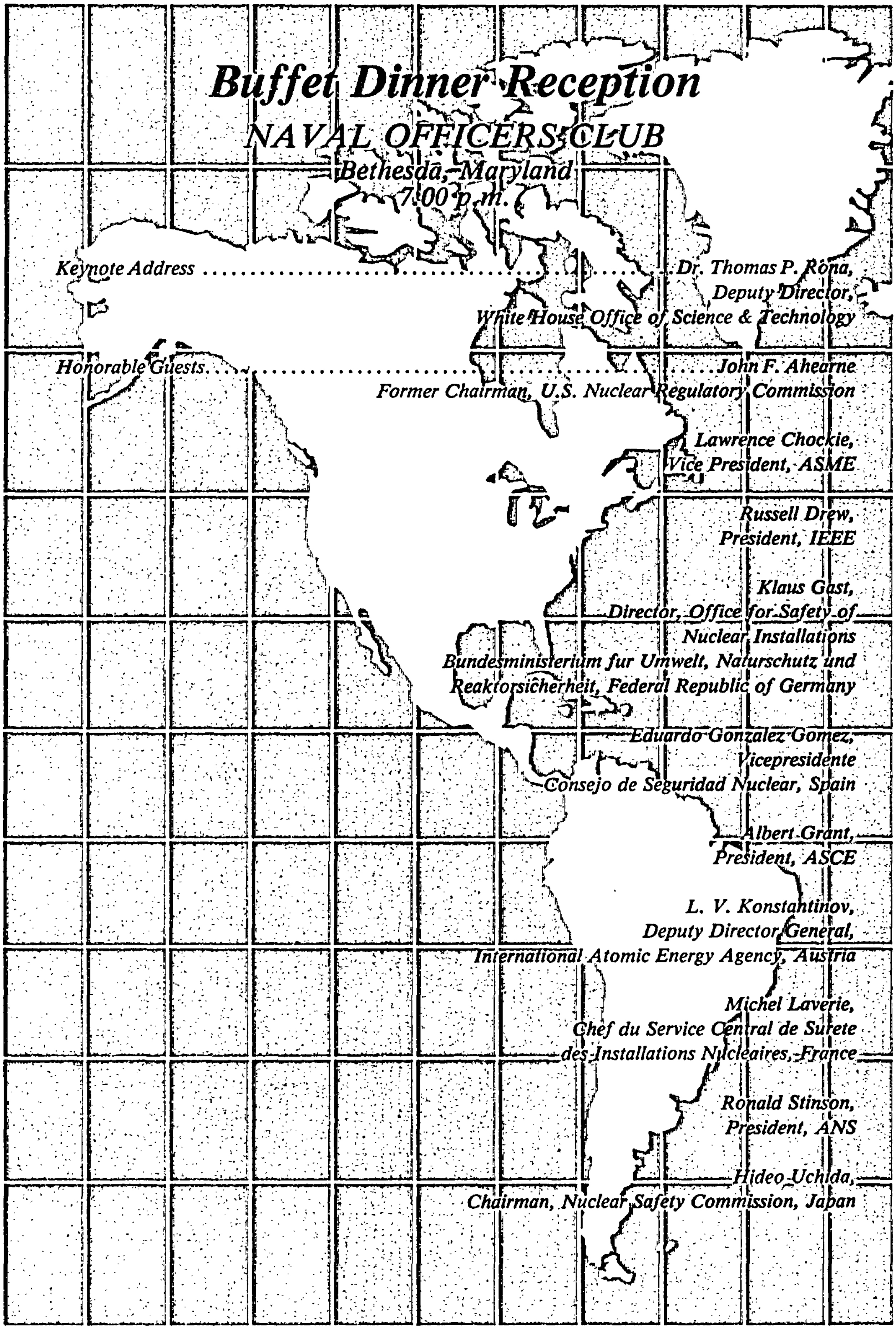
Full-scale applications of the solvents have been very successful. Table 7 lists the cleanings performed to date and the amounts of sludge removal. Note that in some cases only one solvent is applied at a shutdown. This is, of course, governed by the utility's perceived needs and time constraints. Note also that the amount of copper removed from German plant steam generators is generally much smaller than that removed from the Spanish steam generators. This is believed to be because the general feedwater oxygen level is very low in the German plants, and the only source of copper is the condenser tubes.

Summary

One of the observed characteristics of plant aging is the gradual deterioration of PWR steam generator tubes by various corrosion mechanisms. Chemical cleaning of steam generators is one procedure that can be used to increase their life expectancy and can ultimately lead to greater plant availability. It is a procedure that has been used widely in Europe and now appears to be gaining acceptance in the United States. The KWU process and the EPRI SGOG process have both been used on operating plant steam generators with good results. The KWU process is more rapid, due primarily to its higher application temperature, and produces almost no galvanic corrosion. The waste may be handled a bit more easily. Either process can be used effectively to slow the steam generator aging process brought on by some corrosion mechanisms.

TABLE 7. KWU CHEMICAL CLEANING EXPERIENCE FOR PWR STEAM GENERATORS

Year	Plant	Material Removed	Quantity Removed (lb)				SG Manuf	Comm'l Operation	MWe
			SG1	SG2	SG3	SG4			
1985	Biblis-B	Cu	1.3	-	-	-	KWU	1/77	1240
1985	Neckarwestheim	Cu	-	-	6.2	-	KWU	12/76	795
1985	Stade	Cu	6.0	8.6	3.5	6.2	KWU	5/72	630
1985	Stade	Cu	5.7	4.7	-	-	KWU		
		Cu	18.8	18.8	10.6	9.0	KWU		
1986	Neckarwestheim	Fe_3O_4	921	-	-	-	KWU		
1986	Almaraz 1	Fe_3O_4	1012	1197	882	-	W	8/81	930
		Cu	220	247	234	-	W		
		Cu	128	128	106	-	W		
		Fe_3O_4	1060	1345	910	-	W		
1987	Grafenrheinfeld	Cu	-	1.5	-	-	KWU	11/78	1235
1987	Stade	Fe_3O_4	-	-	860	-	KWU		
		Fe_3O_4	-	1389	1389	-	KWU		
1987	Almaraz 2	Fe_3O_4	906	1012	888	-	W	12/83	930
		Cu	79	88	99	-	W		
		Cu	256	306	225	-	W		
		Fe_3O_4	1219	1239	1539	-	W		
1987	Asco 2	Cu	496	644	562	-	W	6/85	880
		Cu	88	66	86	-	W		
1987	Asco 1	Cu	1239	1142	1045	-	W	12/84	880
1988	Almaraz 1	Cu	66	57	93	-	W		
1988	Ringhals 3	Fe_3O_4	744	794	943	-	W	4/81	915
1988	Almaraz 2	Fe_3O_4	1617	1624	1530	-	W		
1988	Almaraz 2	Cu	74	78	75	-	W		
1988	Almaraz 1	Fe_3O_4	1228	1162	1061	-	W		
1988	Almaraz 1	Cu	5.1	9.6	44	-	W		
1988	Ringhals 4	Fe_3O_4	(data not yet available)				W	11/83	915



Buffet Dinner Reception

NAVAL OFFICERS CLUB

Bethesda, Maryland
7:00 p.m.

Keynote Address .... *Dr. Thomas P. Rona,*
Deputy Director,
White House Office of Science & Technology

Honorable Guests .... *John F. Ahearne*
Former Chairman, U.S. Nuclear Regulatory Commission

Lawrence Chockie,
Vice President, ASME

Russell Drew,
President, IEEE

Klaus Gast,
Director, Office for Safety of
Nuclear Installations
Bundesministerium fur Umwelt, Naturschutz und
Reaktorsicherheit, Federal Republic of Germany

Eduardo Gonzalez Gomez,
Vicepresidente
Consejo de Seguridad Nuclear, Spain

Albert Grant,
President, ASCE

L. V. Konstantinov,
Deputy Director General,
International Atomic Energy Agency, Austria

Michel Laverie,
Chef du Service Central de Surete
des Installations Nucleaires, France

Ronald Stinson,
President, ANS

Hideo Uchida,
Chairman, Nuclear Safety Commission, Japan

September 1, 1988

OPENING REMARKS

Satish K. Aggarwal
General Chairman

During past two days, I have stated several times that the nuclear power industry will not survive even a single nuclear power plant accident. The seriousness of the accident hardly matters. This morning, there are two basic questions in my mind, and these are:

Is there a scientifically valid design for an experiment, including theory, research work, and gathering operational data, likely to result in a high-confidence-level prediction for safe operation of aging nuclear power plants? Will it allow the definition of rational, precautionary, or remedial action for extending the useful life of the current generation of reactors?

The second question I have: Are all available resources world-wide properly marshalled toward an effective evaluation of this issue? Should we do anything to lend additional support?

I am hoping that the panel session today will address these two questions.

An attempt should also be made to develop an international policy to deal with aging of nuclear power plants. I also trust that the outcome of this Symposium will prompt the participants to seek intensified participation by universities, utilities, privately funded research centers, government labs, and the government, both within the United States and among our foreign counterparts.

At a minimum, cooperative efforts must be undertaken jointly by the United States, France, Japan, Germany, and Spain. Mr. Laverie, Dr. Uchida, Dr. Gast, and Commissioner Gonzales, I am counting on you to take this message to your governments for international cooperative efforts. It is my hope that you will seek appropriate funding and participation by your countries.

And you, Dr. Konstantinov, I am hoping that IAEA will join in these international efforts. As I see it, international cooperative efforts can only bring success.

I would also like to suggest that the International Nuclear Power Plant Aging Symposium be held every two years, and be rotated among France, Japan, Germany, Spain, the United States -- not necessarily in that order -- until we have obtained all the answers.

Let me just make one more remark. The conditions for success are clearly in your hands.

WELCOMING ADDRESS

Dr. Thomas Murley
Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

It is my privilege to welcome you this morning to the third day of this excellent Symposium. Nuclear power plant aging is, of course, an important safety subject, and we are pleased there are so many world experts here and so many good papers being presented this week.

As the person in the Nuclear Regulatory Commission delegated the licensing responsibility for over 110 nuclear power plants in the United States, I am particularly sensitive to the issues of plant aging. Our first priority, of course, is to be sure that the 110 licensed reactors are operated safely.

With that in mind, I have some advice to you researchers in the audience regarding priorities for aging research. Your first priority, it seems to me, should be to look carefully for new aging phenomena in nuclear plants. That is to say, please search for any new areas of equipment degradation that may have been overlooked up to now. I hardly need remind you that the industrial nations of the world have an enormous stake in nuclear power, and we cannot afford to be surprised by new safety problems as our plants get older.

Once we understand the issues associated with aging, we can take regulatory action to ensure continued safe operation. But if new aging phenomena continue to be revealed from operating experience, it not only impacts the reliability of a large number of operating plants, but it also hinders public acceptance of nuclear power safety.

In the longer term, we must also begin to prepare the regulatory basis for extending the operating life of current reactors beyond their authorized 40 years; perhaps for an additional 20 years.

Therefore, I have a second request for you researchers in the audience. That is, please provide us with a thorough technical understanding of the known aging phenomena and a sound technical basis for making the safety judgments we must make to support plant life extension. Here again, the industrial nations have an enormous financial investment in the current operating reactors.

Turning now for a moment to the topic of this morning's technical session, there is no more important component subject to aging than the reactor pressure vessel. With your indulgence, I would like to take a few minutes to tell a story about the issue of pressurized thermal shock.

It is particularly fitting that the chairman of this morning's session is Mr. Milton Vagins, for it was Mr. Vagins who was the key person on the NRC Research staff who insisted that pressurized thermal shock must be thoroughly studied and dealt with.

This story begins over 10 years ago. At that time, I was responsible for NRC's Reactor Safety Research Program. Most of our attention was focused on loss-of-coolant accidents and emergency core cooling system research, particularly on the large-scale testing facilities like LOFT. We also had, however, a modest pressure vessel research program primarily at Oakridge National Laboratory, and it was doing very good work.

You may recall that in the mid 1970s, we had received a nasty surprise. Surveillance specimens were showing higher than expected neutron embrittlement of some weld materials. It was found to be due to the copper impurities in some welding rods. And parenthetically, I should add that this is the kind of surprise that we hope that you researchers will prevent in the future.

At that time, NRC took regulatory steps to protect against this problem by setting operating limits and requiring equipment to prevent low-temperature overpressurization. We thought we had solved the problem.

But then Mr. Vagins came to me and started telling of this problem called pressurized thermal shock. I believe this was in 1979. I did not understand what he was talking about. He patiently explained how there could be severe overcooling transients at full system pressure where thermal stresses and hoop stresses were superimposed and could potentially cause crack initiation if there were a flaw present in one of the beltline welds of a pressure vessel.

Well, that was significant news indeed. We then redirected NRC's research program to better analyze this issue and to conduct some scale model vessel experiments.

At first our friends on the licensing staff and in the industry were skeptical that this was a real problem. It was as if we were unwelcome guests at a dinner party. But Mr. Vagins was persistent, and there was no escaping the facts. In 1978, one of our new reactors, Rancho Seco, had experienced a severe transient in which the vessel was cooled down some 160 degrees Centigrade in one hour coincident with full system pressure. Calculations by our research staff showed that if this transient were to occur at the end of life, and if a weld flaw were present, the consequences could be serious.

Unfortunately, at that time we in the NRC were all distracted by the TMI-2 accident for nearly a year. But in 1980, we turned to give primary attention to resolving this issue of pressurized thermal shock.

About that time, we also began to receive another surprise on the importance of nickel in weld metal, as well as copper, in its impact on embrittlement.

But you all know that this story has a happy ending. The NRC and the industry worked closely together to resolve the issue. We developed a rule that we now believe provides adequate protection against pressurized thermal shock.

There is a lesson in this story, and it is, to strengthen my advice to you researchers, to please be thorough in looking under all the rocks and in all the nooks and crannies for any new aging phenomena.

PRINCIPAL ADDRESS

Michel J. Lavérie

Director of the Central Service
for the Safety of Nuclear Installations

Ministry of Industry - France

Significance of French Standardized Design in Managing Aging in Nuclear Power Plants: The Point of View of the Safety Authorities

Safety decisions concerning power plant aging constitute an especially difficult problem as:

- First, these decisions often involve a painful confrontation between safety and economic motivations,
- Second, the technical file, which is the basis on which a decision is reached, is never either completely white or completely black. Generally speaking, it is impossible to pinpoint an exact day when power plant safety switches abruptly from an acceptable to an unacceptable state. The selected date is always, at least partly, arbitrary.

It is extremely difficult, and even well nigh impossible, to determine in advance the decommissioning date for a reactor, which implies assessing a priori the period during which it will operate satisfactorily. However thorough may be the behaviour analysis of a type of component, however detailed may be the theoretical analysis of various incidents or events liable to make a component more vulnerable or lower its resistance, they can never replace precise concrete assessment of operating conditions in each plant.

The hard and fast safety rule must be that at any time, and not only after a certain period, the operating utility must be in a position to prove that a satisfactory safety level is assured in each reactor. If this is not the case, the corresponding plant must be shut down, whatever may be its forecast lifetime.

In practice, the work methods implemented vary greatly depending on whether the power plant is of a special type or is one of a standardized series.

It should be borne in mind that 7 nonstandardized power reactors are currently extant in France:

- 4 reactors of the natural uranium-graphite/gas series
- 2 reactors of the fast neutron series (Phenix and Creys Malville)
- 1 reactor of the PWR series (Chooz A)

There are 47 standardized PWR units in two main series:

- 34 units in the 900 MWe series
- 13 units in the 1300 MWe series

Also, a dozen units are currently under construction which belong to the 1300 MWe series just referred to, and the most recent ones belong to a new plant series called N4 with an energy output of 1450 MWe.

In order to illustrate the advantages of standardization in monitoring these reactors, I will compare the practices implemented for nonstandardized reactors with those for standardized reactors.

1. Nonstandardized Reactors

No purpose would be served by summarizing once again the already well-known authorization, construction, and startup procedures for a reactor of this type. I will simply describe the methods by which safety is reassessed throughout the entire life of a reactor.

1.1 Systematic Safety Reassessment

Safety reassessment is undertaken on the operator's own initiative or at the explicit request of the safety authorities each time it seems likely that there are lessons to be learned from an event which has occurred at another installation or in order to benefit from a significant gain in experience and knowledge. Hence, to give some concrete examples, each French reactor was the subject of reassessment in light of the fire in 1980 at the reprocessing plant at La Hague and the Three Mile Island and Chernobyl accidents. Reassessments of this kind are partial with respect to the common points between the subject under consideration and the installation characteristics.

Also, a complete safety reassessment is programmed periodically for each reactor. Periodic assessments are currently programmed at 10 years. At the end of this assessment, the safety authorities notify the operator of additional conditions for continued operation.

Two different types of questions arising out of these reassessments should be distinguished:

- Does the installation still have the same safety level as was assumed at the commissioning stage? If not (either because the safety level has deteriorated or because the initial risk factor was underestimated), it is incumbent upon the operator to remedy the situation.
- If the installation does not meet the most recent safety standards, how can the divergence in safety level be quantified and how can this divergence be overcome within the limits of technically and economically reasonable actions?

An inflexible rule of initial safety level should be maintained and to this rule can be added, on a case-by-case basis, more recent safety objectives.

To illustrate this approach, I will give an example taken from the 10-year assessment of the natural uranium-graphite/gas reactors at St. Laurent:

- Reliability of the CO_2 cooling function was shown to be rather lower than expected. The operator was obliged to specify the required measures for bringing the reactor back into line with its initial reliability forecast.
- The plant did not comply with the most recent rules concerning earthquake resistance. The operator determined the existing margins in liaison with the safety authorities on a case-by-case basis and investigated the possibility of specific reinforcements.

1.2 Evaluations Relating to Problems Specific to the Reactor Under Consideration

Aging of certain components can lead to major modifications. These modifications require a new demonstration of a satisfactory safety level and are the subject of regulatory authorizations concerning resumption of operation.

A situation of this kind can also lead the operator to opt for definitive reactor shutdown. But this decision is generally based on a complex set of factors. For example, two older reactors were recently shut down in France: Chinon A3 (natural uranium-graphite/gas series) and Brennilis (heavy water reactor). In both cases, specific safety problems had arisen, but this aspect was only one element of the decision among other, notably economic, considerations.

Recently, three French reactors have encountered serious technical problems with consequences for safety which cast doubt upon the advisability of continued operation. These reactors are:

- Chooz A, a PWR 310 MWe reactor for which the risk of reactor vessel fracturing under irradiation has required significant investigation.
- Chinon A3, a natural uranium-graphite/gas reactor for which the corrosion of internal support structures has required long repair sequences performed by maintenance robots.
- Bugey A, a reactor of the same series for which doubts have been raised concerning resistance of the graphite stack.

I have considered it useful to present the types of problems encountered and the way these problems were dealt with in a more detailed manner in Appendix 1.

1.3 Handicaps

Monitoring of nonstandardized reactors, even if it is as thorough as could be reasonably hoped, has two major handicaps by comparison with standardized reactors:

1. Operating feedback is limited,
2. Design and analysis studies are also limited, from the point of view of the operator as well as that of the safety authority.

Clearly, the same resources cannot be implemented for an isolated reactor as for an entire standardized series of reactors.

2. Standardized Reactor Series: The Strengths of Standardization

2.1 Analysis Methodology

A standardized reactor series with several tens of units has two fundamental advantages: the breadth of operating feedback and the focusing of safety considerations on a single reactor type.

It is possible to work in a more continuous fashion in order to improve basic safety. Hence, the 34 units of the French 900 MWe series are all subject to four analysis procedures:

1. Analysis of all events (about 500 per year on this series) designated "safety-related" and finalization of contingent corrective measures on all relevant units (staggered over a period of time. Studies are undertaken progressively at the stimulus of events; corrective measures are taken as soon as technically feasible and prioritized with respect to technical importance).
2. Analysis of modifications to be performed so that the development of safety doctrine can be taken into account. Even if regulatory practices defined for the most recent reactors cannot be made automatically retroactive, the operator is systematically requested to investigate existing divergencies and "technically reasonable" methods for reducing this divergency (staggered over a period of time on a case-by-case basis).
3. Overall investigation of operating feedback from the 34 units, enabling broader, more comprehensive lessons to be learned than would be possible by individual investigation of each significant event (every 2 years).
4. Reassessment of the safety of each unit (every 10 years).

The methodology which has just been described has two simultaneous but clearly distinct objectives:

1. Safety improvement by capitalizing on experience and development of know-how,
2. Detection and remedy of signs of aging in safety-related equipment with, if possible, adequate provision being made for this eventuality in advance. It is this latter point which I intend to discuss today.

2.2 Allowance for Aging

How should reactor safety reassessment procedures be performed for the correct management of aging problems? A more detailed account of this subject is given in Appendix 2.

2.2.1 Design of Critical Components

A first step would be to examine all equipment known to be "short-lived." A specific research program must then be implemented concerning these components to determine their probable lifetime and, from a safety standpoint, to provide means of appraising their exact condition so that they can be removed in time.

The purpose of this program is to enable a "fit for duty" appraisal to be awarded for each item of critical equipment. Thus, as for the steam generator tube bundles, at the end of each unit shutdown, the defects revealed by eddy current inspection and the laboratory results obtained on tubes removed for tests are considered and an appraisal issued as to whether or not the unit should be authorized to restart.

"Alarm bell" systems must also be provided. The various primary system structural materials are designed to withstand a certain number of situations with a significant safety margin. The fact that this number has been reached by no means implies that the reactor must be shut down. Reaching the design basis number of situations is simply a kind of "alarm bell" indicating that care must be taken to organize more thorough analysis to determine how much longer the equipment under consideration can continue to operate.

The same approach is adopted for the reactor vessel. Determining the age of a vessel on the basis of the number of neutrons received having an energy exceeding 1 Mev or 1 Gev can only indicate an alarm threshold. It can provide no accurate information concerning the actual condition of the vessel.

2.2.2 Probabilistic Risk Assessment

PRA might seem to be an attractive approach as it provides an overall view. Unfortunately, the enormous uncertainty of analyses of this kind runs the risk of completely obscuring those problems specifically connected with aging. On the other hand, these studies have the undoubted advantage of highlighting the weaknesses of certain components, whose failure rate weighs heavily on the final result.

2.2.3 Incident Analysis: Judgment, Periodic Tests, and Maintenance

All significant incidents are analysed by three parties: the local operator, the EDF Central Division, and the safety authorities. This analysis enables a judgment to be made regarding particular problems. Over the longer term, it also enables a more general judgment to be supported regarding not only the quality of the equipment itself, but also the adequacy of periodic tests and maintenance operations to which it has been subject.

2.2.4 Qualification and Aging

This aspect is, obviously, extremely important. As the qualification process is both unwieldy and costly, it cannot be exhaustive. Standardization has enabled this load to be distributed over a large number of reactors and to concentrate on standardized equipment, and in this way the largest number of results has been made available.

3. Standardized Reactor Series: Potential Standardization Difficulties

3.1 Management of Operating Feedback

A standardized plant series provides an increased level of operating feedback for each of the units comprising the series. Weaknesses are more easily detected and, consequently, a much greater number of weaknesses, the existence of which would have remained unknown in a single nonstandardized reactor, must be corrected.

In concrete terms, this means that for the 34 units of the 900 MWe series, thousands of modifications are performed. The design and performance standard of these modifications must be faultless, as nothing could be worse than a badly implemented improvement.

3.2 Confronting all Major Problems Without Immobilizing the Plant Series

It is unthinkable that all 34 units of a single series be abruptly shut down or maintained in a shutdown state for a long period of time. The only way of avoiding this situation is to make provision sufficiently early for problems which might seriously affect continued safe operation. However, particular caution should be exercised in this area. This method implies an extremely pessimistic analysis of any potential problems which might develop in an unfavourable manner.

All difficulties must be anticipated sufficiently early to facilitate elaboration of a solution in parallel with continuing plant operation and to enable any required maintenance operations to be spread out over a period of time.

A current illustration of this kind of situation in France is the sometimes rather worrying crack development rate of steam generator tubes. It is vital that action be taken before a situation liable to compromise safety occurs. Certainty regarding repair work must be obtained as quickly as possible for steam generator alterations and also concerning availability of corresponding equipment. In certain units, these repairs must also be performed earlier than required so that maintenance work on the entire series can be spread out over a period of time. Repairing five reactors annually - the same rate as reactor commissioning - would, in fact, be very difficult.

For these reasons, maintenance operations are actively prepared in advance. But can it be asserted that all generic "illnesses" for this reactor series will leave those responsible a sufficient time period for implementation of repair work without serious disturbance of availability?

4. Conclusion

Standardization provides many advantages for dealing with power plant aging problems: wider experience and more concentrated action. It also enables more subtle and longer lasting follow-up actions.

On the other hand, standardization involves anticipation of faults in order to be certain of forestalling problems. This is a wager that both the operators and the safety authorities will do their utmost to win.

APPENDIX 1

Nonstandardized Reactors
Three Examples of Reactor-Specific Problems

1. Chooz

1.a Description

The nuclear power plant built in the Ardennes region belongs to the Société d'Energie Nucléaire Franco-Belge des Ardennes (SENA), in which Electricité de France (EDF) and a group of Belgian utilities each have a 50 percent holding. This 310 MWe plant, first connected to the grid in April 1967, is equipped with a pressurized water reactor with four reactor coolant loops.

Following systematic safety review performed as early as 1983, operation was authorized to proceed, providing certain safety improvements were made concerning the safety injection system, the steam generator auxiliary feedwater supply, and the control room.

1.b The Problem

In 1987, tests performed on representative irradiated specimens taken from the vessel (during the shutdown in 1986) appeared to indicate faster deterioration than expected of vessel mechanical properties. In August 1987, on the basis of the dossier submitted by the operator, it did not prove possible to demonstrate that there was no risk of vessel failure during the next cycle. It was consequently decided not to authorize plant restartup.

In February 1988, a new dossier was submitted, giving details of additional tests performed with a view to obtaining a more accurate assessment of the real state of the vessel (reconstitution of test pieces, sampling of metal filings, nondestructive testing) and of the actual failure margins. Provisions were also made to increase the temperature of the safety injection system water. After examination, it was decided that operation of the plant should be resumed, without, however, exceeding conditions equivalent to 20 years of operation at rated power, with an 80% load factor, save as otherwise provided by appropriate additional justifications.

1.c Restart

Restart was authorized on March 23, 1988, after the operator:

- Acquired a better knowledge of the real extent of core degradation by means of tests and improvements to design models showing that the risk was acceptable, and
- Performed modifications enabling the temperature of the injection water to be raised, hence reducing the amplitude of any possible thermal shocks (by reducing the divergence between the safety injection temperature and that corresponding to an equipment phase transition).

2. Chinon A3

2.a Description

This reactor is cooled by carbon dioxide gas circulating at a pressure of about 30 bar in channels in the stacked graphite moderator enclosing the fuel elements. Chinon A3 started up in 1966 and is still operating at the maximum authorized power level of 360 MWe.

In 1974, owing to large-scale cracking, the main primary system CO_2 nozzles had to be replaced.

During a shutdown in 1980, doubts were cast on the corrosion resistance of the steel structures and upper internals in a hot CO_2 environment. After a further shutdown to assess the extent of the phenomena considered, a temporary operating permit was granted so that methods of repairing these structures could be defined and implemented. The first repair stage came to an end in 1987, after structure reinforcement performed by welding robots, designed for automatic approach and fitup. The repair period was used to make an overall assessment of the plant unit.

2.b The Problem

Since 1970, doubts were expressed regarding the strength of steel structures supporting the instrumentation equipment for the clad failure detection assembly. The risk involved breaking of these structures and loss of too large a number of thermocouples. In 1980, breaks of this kind were detected, and in 1984 it was decided to shut down the unit.

2.c Restart

The unit was restarted in November 1987 after those sections designed for load recovery of the existing supports and holding of component parts in position were welded by means of a remote-controlled robot. The operation was performed by remote-controlled handling devices which utilize robotic technology enabling automatic approach and fitup.

3. Bugey A

3.a Description

This 540 MWe reactor, similar in design to that of Chinon, came on line in 1972. As early as 1974, provisions were made to limit graphite corrosion. Since 1981, the graphite stacks have been subjected to overall monitoring (coring, camera). In 1986, it could be soundly asserted that graphite wear was not a subject of concern.

Current assessments, allowing for all uncertainties, give the reactor a three-year margin. Further coring and renewed attention to monitoring and the measures taken should enable the validity of this judgment to be confirmed between now and the end of 1988.

3.b The Problem

In 1984, following a safety inspection of this power plant, the Central Service for the Safety of Nuclear Installations (SCSIN) requested that the strength of the graphite stack be re-examined. The risk was of partial or total collapse of the column of graphite bricks under its own weight, which could have caused fusion of some parts of the core.

In January 1988, noting that the strength of the graphite had not been proven, the SCSIN informed the operator that shutdown of the unit would be requested on June 30 if an adequate demonstration of graphite stack strength had not been given by that date.

3.c Restart

Before the appointed date, the operator carried out a new modelling procedure of graphite pile wearing based on the graphite coring previously performed. In theory, the new calculations gave an operating margin of three years for the reactor. However, new core drilling carried out at the end of 1988 should enable the validity of this process to be confirmed.

4. Lessons

In each of the cases presented, at a given point in the plant's lifetime, doubts related to the appearance of new elements, little known when the reactor was designed, were cast on its real safety level.

These doubts naturally led the safety authorities to request or envisage shutdown of the plant, the restartup or continued operation of which is conditional upon replacement of the incriminated equipment or improvement of knowledge in a specific area, usually based on more accurate assessment of the exact condition of the equipment, substantiated by elaboration of a calculation model to estimate how the degradation is likely to evolve.

It is certain that during 30 years of nuclear power plant operation we have learned a lot and many safety questions have already been dealt with. But our experience of plant aging is, as yet, fairly slight and we should be wary of over-assertiveness.

Whatever the case, restartup of a plant when there are doubts as to the validity of nonreplaceable equipment can only take place if new elements are put forward enabling these doubts to be refuted.

In the case of Bugey, pieces of graphite were sampled; in the case of Chooz, test pieces representative of the vessel were reconstituted. Whether destructive or nondestructive examination methods are used, it would seem essential to be able to determine the exact condition of the equipment considered.

This requirement has, in fact, already led to a considerable increase in both quantitative and qualitative nondestructive testing in most nuclear power plants. It is doubtless superfluous to recall the vital role played by nondestructive tests in assessing the sub-linear defects of the first French 900 MWe vessels, or in detecting the intergranular stress corrosion cracking in certain BWR piping.

For different cases, analyses must also be based on real physical models, which can be used to predict how the main parameters will develop: buildup rate of the reference temperature for nil ductility transition (RT_{NDT}) in the case of Chooz, or the rate of graphite wear in the case of Bugey.

APPENDIX 2

How is aging to be taken into account during reassessment of units belonging to the 900 MWe reactor series?

1. The Critical Components

A first step would be to examine all equipment known to be "short-lived." A specific research program must then be implemented concerning these components to determine their probable lifetime and, from a safety standpoint, to provide means of appraising their exact condition, so that they can be removed in time.

EDF has already listed 18 "sensitive" items of equipment, i.e., "equipment which cannot be replaced in the context of routine maintenance programs." These include:

- The main NSS components
- The containment
- The control and instrumentation systems
- The electrical networks (cables)
- The turbogenerator sets

EDF has organized a number of discussions and exchanges of views on this equipment in compliance with French statutory requirements whereby, in the first instance, the operator is liable for the safety of the plant. The main conclusion reached at present with respect to safety is that these components should have a lifetime exceeding 40 years inside the plant, except for certain sets of equipment such as the control and instrumentation system or the steam generators which can be replaced.

One of the objectives of this study is to enable a "fit for duty" rating to be awarded to each critical item of equipment.

Thus, as for the steam generator tube bundles, at the end of each unit shutdown, the defects revealed by eddy current inspection and the laboratory results obtained on tubes removed for tests are considered and an appraisal issued as to whether or not the unit should be authorized to restart. This appraisal, the terms of which are usually evident, sometimes requires the removal of 20 or 30 tubes and the elaboration of calculation models from which degradation developments can be derived.

This notion of a "fit for duty" rating would appear to be essential, but it should not be confused with another notion, which corresponds to an alarm signal. The various primary system structural materials are designed to withstand a certain number of situations. The fact that this number has been reached by no means infers that the reactor must be shut down. Reaching the design basis number of situations simply indicates that care must be taken to organize more thorough analysis before issuing the "fit for duty" rating. In this particular case, the equipment could be subjected to inspection involving special instrumentation, such as fatigue monitoring modules.

Such a system, which has already been described on many occasions, should enable partial and cumulated aging to be permanently monitored for the critical areas of the primary system.

The problem is, in fact, the same for the reactor vessel. Determining the age of a vessel on the basis of the number of neutrons received having an energy exceeding 1 Mev or 1 Gev can only indicate that precautions should be taken. It can provide no accurate data concerning the actual condition of the vessel.

This notion is apparently not yet applied to control and instrumentation. For the operating utility, "complete replacement of electronic equipment shall be envisaged after a period of about 25 years, on grounds of technical and economic obsolescence." Failure of certain control and instrumentation equipment prior to this deadline would nevertheless lead to operation with a combination of new and old components, which raises the following two questions:

- To what extent would compatibility between these two categories of equipment call in question the safety level?
- To what extent would an increase in the failure rate be acceptable?

In any case, this approach cannot be considered as comprehensive since, even if simple measures are implemented enabling components to be rated "fit for duty" or not, such measures will only apply to a limited number of components.

2. Probabilistic Risk Assessment (PRA)

Probabilistic safety studies, on the other hand, apply to all components equipping a power plant and should thus enable the critical components to be determined.

It is not irrelevant to recall, however, that PRA results are to be handled with caution. Because human errors are not considered, because equipment failure values always have a strong arbitrary strain, and because the basic postulates differ from one assessment to another, it is obvious that these figures cannot be taken as absolute values.

Similarly, the idea of applying to a power plant the PRA results obtained for another plant seems to me hazardous, unless the plant units considered are exactly identical from both design and operation standpoints. Experience with these studies shows that there is little in common between 900 MWe and 1300 MWe PRA results.

On the other hand, these studies have the undoubted advantage of highlighting the weaknesses of certain components, whose failure rate weighs heavily on the final result.

In the same way, if PRA studies are resumed after an interval of several years, they should enable incipient weak points to be detected. However, interpretation of PRA-evidenced parameter changes is a delicate operation. Are the changes due to aging, to equipment modifications, to a different operating mode, or even to a new calculation method?

3. Incident Analysis

The safety authorities are informed of all significant incidents, each described in a report which in France is analyzed by three parties: the local operator concerned, the EDF central division, and the safety authorities. Operating feedback should consequently be integrated into safety reassessment both to check the incidence on a given plant of the lessons derived from the various relevant analyses and to examine in a new light all the incidents which have occurred with a view to re-analyzing some of them which may have seemed too exceptional to be of interest to the first analyst, but which could subsequently prove to be generic.

Moreover, the inclusion of operating feedback in a safety review should enable questions to be approached from a slightly different angle: the operator would not be required, in this case, to find solutions to specific problems, but to use the operating data accumulated over several years to reconsider more general questions concerning the validity of the periodic tests, the maintenance programs, or even various items of equipment.

For some, multiplying by ten the number of periodic tests performed on a component can but enhance safety. Paradoxically, this is not at all the case if the failure rate induced on other equipment by the special test conditions imposed increases by a factor of 100. This problem can only be solved by an overall analysis based on several years of operation and accurate equipment background data.

Finally, in the context of standardized plants, operating feedback in the safety reassessment of a given plant unit should enable particular trends of the unit to be defined, which could prove a valuable source of data on the whole series of plants.

4. Qualification and Aging

With the exception of equipment failures under accident conditions or during periodic tests simulating accident transients, what we have so far discussed is incomplete in that these analyses throw little light on equipment behavior under accident or post-accident operating conditions.

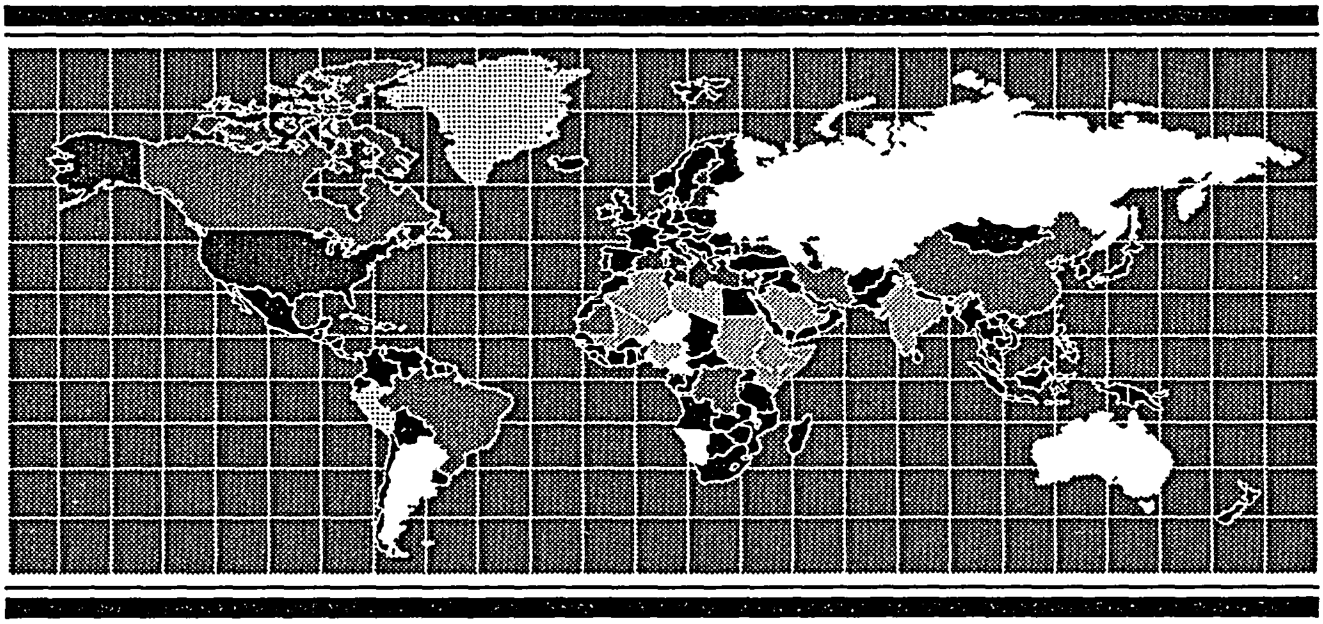
It must be remembered that, in France, the operator has to submit an exhaustive list of components required to operate under accident and post-accident conditions, as foreseen at the design stage, to define for each component the most penalizing conditions under which it will be required to operate and to demonstrate that each will withstand the operating conditions previously defined.

In addition, for equipment required to withstand an earthquake or operate under conditions corresponding to those prevailing in the containment after a loss of coolant accident (LOCA), according to French regulatory practice, aging tests must be performed beforehand in order to obtain the best possible assurance that the equipment will operate satisfactorily after 20 or 30 years of service in a nuclear power plant.

This particularly time-consuming and costly qualification procedure could only have been implemented in France in the framework of the standardized series policy. Components which, in other circumstances, would be produced as single items, here represent a production series from which one or several specimens can be removed for test purposes.

There should be a reassessment procedure to check that equipment operating on the site ages in the same way as in the qualification tests, from which it could be concluded that its operation under accident conditions would be the same as during the qualification process. If this is not the case, the qualification should be called into question.

This process for qualification ratification after a period of time, possibly leading to annulment, may seem extremely costly and unwieldy to the operating utility, but the purpose of such a process is one of the aims of a safety review.



TECHNICAL SESSION 7

Aging of Vessels and Steam Generators

September 1, 1988

Session Chairman

MILTON VAGINS
Chief, Electrical and Mechanical Engineering Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

PTS EVALUATION IN FRANCE : THE CASE OF CHOOZ A REACTOR

F. HEDIN
S. BAUGE
J.C. GUILLERET
B. BARTHELET

SUMMARY

Reactor Pressure Vessel material degradation and ageing due to neutron irradiation is of utmost importance with regard to the safety and reliability of a reactor vessel during events like pressurized thermal shock.

After presenting the basic methodology used in France for PTS evaluations, this paper will expose the case of the CHOOZ A reactor, which recently raised important questions that were to be answered after extensive investigation work.

Considering that :

- uncertainties on RPV and specimens fluence evaluation were significant,
- from the two forged ring materials (B and C) constituting the beltline region, only the C-Ring material was present in the surveillance program and the B ring Phosphorus-content was higher than the C-Ring one,
- examination of the beltine region, conducted in 1986 for other reasons, were difficult to interpret,
- therefore B-Ring RT_{NDT} value and associated uncertainty were too high,

the French Safety Authorities did not allow the restart of the reactor without achieving some complementary investigations.

These investigations were conducted from June 1987 to February 1988 and consisted in :

- reevaluating fluence by means of a tridimensional neutronical Monte Carlo code and direct cladding cuttings measurements,
- reconstituting B-Ring from irradiated HAZ specimens to get a direct approach of the B-Ring behaviour under irradiation,
- making non-destructive examination of a part of the beltline region.

The results led to a mean end of life RPV - RT_{NDT} of 80° C, and provided satisfactory margins with respect to the PTS evaluation, with a safety injection temperature of 50° C. The plant was allowed to restart in march 1988 and can operate without reevaluation insofar as it does not exceed a fluence corresponding to a cumulative life of at least 20 years at nominal power with a 0.8 lead factor.

PTS EVALUATION IN FRANCE : THE CASE OF CHOOZ A REACTOR

Determination of margins in relation to the fast fracture of the vessels of operational pressurized water reactors is of vital importance to the reactor's safety.

After presenting the evaluation method for fast fracture of vessels in FRANCE, we intend to discuss the case of the CHOOZ A power station, which is the oldest PWR power station in operation in FRANCE, and for which the behavioural justification to fast fracture resulted in important work being carried out during the period June 87 to the first quarter of 1988.

EVALUATION OF MARGINS IN RELATION TO FAST FRACTURE

The decree of the 26th January 1974 relative to the regulation of pressurized equipment of nuclear steam supply systems requires that "the Constructor shows that there is no fast fracture risk with the equipment" and that the Operator makes sure that this prescription is permanently satisfied.

Up to this moment in time, no regulatory criteria has been defined of the screening type of criterion (1) existing in the UNITED STATES based on the RT_{NDT} of the vessel, in order to evaluate the behaviour of vessels in operation.

The traditional design approach (code RCC-M annexe ZG) is essentially to consider an initiation criteria (K_I/K_{IC}) of the rupture, under normal, disturbed and 3rd category conditions, the possibility to allow for the arrest offered in 4th category conditions, the margins depending on the category of the loading condition. The conventional defect, a semi-elliptic one of about 15mm depth through the cladding, is analysed according to the linear elastic fracture mechanics. A possible alternative approach consists of studying the behaviour of a defect with a depth equal to 1/4 of the shell thickness and to compare K_1 to K_{1R} (cf. code ASME).

The approach followed during operation consists of :

- limiting uncertainties as much as possible, and utilizing the results of the irradiation surveillance programme constituted by test specimens representative of the vessel material and by the internally associated dosimetry,
- using the results of the in-service vessel control, in particular the zone located below the vessel lining,
- supplying a reliable calculated evaluation of the margins in relation to fast fracture, by verifying that explicit margins compared to the material's toughness are of the same level as those justified in the design.

This analysis is made on the basis of the fast fracture behaviour in accidental conditions of réference defects defined from manufacturing and control conditions of the vessel :

- for the circumferential weld of the core shell, the circumferential orientation envelope defect corresponding to the maximum size of the technologically envisaged defect is taken as equal to 6mm, and for an exceptional defect, corresponding to a size which is very unlikely that it goes undetected by the controls, is taken as equal to 12mm ;

- for the forged shell, the size of the longitudinally oriented reference defect is taken as equal to 4mm (envelope) and 6mm (exceptional).

The study concerns essentially under-clad defects and is completed by the study of a conventional defect through the cladding.

The transients have been the subject of a specific analysis related to the fast fracture risk, and this is particularly valid for small primary breaks of 1" or 2" in the hot leg, classed in 3rd category condition, leading to a stagnation of the primary flow and which are enveloped for this category of condition ; LOCA is the envelope loading studied in the 4th category condition.

The admissibility of margins is thus essentially justified by calculation concerning the risk of the rupture starting for the reference defects, the margins depending on the probability outcome of the defect/loading combination (cf. table 1).

They are mainly linked to the $RT_{NDT} - T_{IS}$ difference (T_{IS} is the safety injection temperature) (cf. Table II) : in particular, these results enable the minimum in operation temperature of the safety injection water to be determined.

EVENTS SEQUENCE AND MAIN CHARACTERISTICS OF THE CHOOZ A VESSEL

The CHOOZ A power station was commissioned in 1967, and the power has been progressively increased reaching a value of 1040 MWth in 1975. The cold leg temperature varied from 257°C to the current value of 265°C.

The CHOOZ A vessel was built in 1963/64 of forged core shells made from a low manganese-nickel-molybdenum alloyed steel (1.2 MD.0.7. grade A). The core zone consists of part of shell B and C and the circumferential weld B/C (cf. fig.1).

The chemical composition of the two shells and of the surveillance programme test specimens are similar, the difference mainly concerning the phosphorus content (0.005%) (cf. table III). The RT_{NDT} values in the pre-irradiation state, given in table IV, were estimated for the welds, from the Charpy V resilience transition curves and for the base metal, from the Pellini and Charpy V tests.

The surveillance programme of the effects of irradiation of this vessel was made, following the modification of the internal structures carried out in 1968/1970, by using containers located under the lower core plate (cf. fig.2) containing the test specimens on two levels and dosimeters on three levels, in a zone with an important axial flux gradient.

The programme consists of 8 containers placed in 1970, two others having been foreseen in reserve (cf. table V) ; one was inserted into the reactor in 1975, and the other was used in 1982 to verify the initial resilience properties.

The surveillance programme consists of the Charpy V resilience test specimens :

- parent metal of shell C sampled in the length direction between 1/4 and half thickness,

- built up metal from a weld not entirely representative of the actual B/C weld, made from a cutting of shell B, the notch being located either in the weld, or in the HAZ (Heat Affected Zone) (fig. 5),

- reference metal (plate in A 302 B).

The surveillance dosimetry consists of copper dosimeters. A complementary dosimetry internal (Ni, Cu, Fe, Nb, Cu/Co, U, Np) and external (Fe, Ni, Al/Co) to the vessel was carried out during the 12th cycle of the reactor from September 1982 to August 1983.

Margins with respect to vessel rupture were re-evaluated in 1983, including an in-depth study of the most severe transients concerning this type of damage.

The maximum fluence of the vessel at the design end of life (20 years, 80% availability) was evaluated at that time at 3.4×10^{19}.n.cm^{-2} (E > 1MeV). The irradiation embrittlement of the core zone materials were estimated using the "FIS" envelope formula, established for 34 900 MW EDF vessels (ref. 3), increased by a factor of 1.4 to allow for the lowest operating temperature of the CHOOZ A. The margins in relation to the risk of vessel rupture were thus evaluated in 1983 for an RT_{NDT} end of life of 95°C maximum, associated with the following dispositions : heating the safety injection water to 50°C, modification of the primary pumps shut-down criteria.

NEW FACTORS WHICH APPEARED IN 1987

The utilization of the surveillance test specimen capsule 6 in 1986 completed by capsules 4 and 9 resulted in the Operator modifying his embrittlement forecasts for core zone materials : indeed the value of the RT_{NDT} shift for shell C measured for a fluence of 3.18×10^{19} n.cm^{-2} (E > 1MeV) was 106°C (cf. fig 3).

The provisional value of this shift, deduced from various formulae (FIS, RG 1.99.1, GUTHRIE,...) was about 90°C ; the difference with the measured value was difficult to explain just on the basis of temperature. Investigations were carried out on the irradiation conditions of this capsule, and the analysis of the results could not explain this result.

Other materials of the control programme (HAZ, weld, reference metal) showed a lower irradiation sensitivity (fig. 3).

Finally, and though it is very likely that the RT_{NDT} of the vessel does not have an excessive value, probably not more than 80°C to 100°C, the Safety Authorities required that an important work programme should be carried out before restarting the plant, in order to appreciably reduce uncertainties on the RT_{NDT} of the vessel.

Indeed there was the fear that shell B, which is not checked in the surveillance programme, can have a more severe embrittlement than shell C, because of its higher phosphorus and its rather low copper contents.

The work programme which was carried out is given below.

COMPLEMENTARY WORK LINKED TO RESTARTING THE CHOOZ A POWER STATION

Important work was carried out between June'87 and March'88, to enable the CHOOZ A plant to be restarted :

- a re-evaluation of all the neutron properties on the basis of design and direct activity measurements,
- determination of the shell B embrittlement with the help of reconstituted test specimens,
- non-destructive control of part of the vessel's risk zone under the cladding.

Neutronic Re-evaluation

A complete re-evaluation of neutronic properties specific to the CHOOZ A vessel was carried out in order to minimize uncertainties by using neutronic power analysis, for comparing the calculated results to the measurements made during a special dosimetry, and by carrying out a direct activity measurement on cuttings of the vessel's cladding and on the surveillance programme test specimens.

After a critical examination of the neutronic data required for the calculations, the uncertainty associated with the power of the peripheral assemblies accrued during 15 operating cycles has been estimated at about 5% ; the reference neutronic spectrum calculations were made with the help of the TRIPOLI 2 code, using the MONTE-CARLO method, in 3D for the surveillance test specimens and 2D for the vessel.

For the Main Vessel :

- The comparison between the calculated and measured flux for the special in-cavity dosimetry cycle 12 shows a relative deviation of + 12% for the Fe54 dosimetry and
  + 15% for Ni58.
- The end of life calculated fluence (20 years, load factor 80%) for a power of 1040 MWth is 4.6 x 10^{19}.n.cm^{-2} (E > 1 MeV), which is about 35% more than previous evaluations, the maximum associated uncertainty being ± 11% (table VI).
- Activity measurements on two cuttings of the stainless steel cladding of the vessel, one from the core median and the other at the B/C weld level utilized on the Mn54, gave an overestimation of the calculated fast neutron flux compared to the measured flux of about 10% (table VI).

For the surveillance test specimens :

- the comparison between calculations and measurements made during special dosimetry shows good correlation, the mean dispersion being significantly less than 10%, a deviation of 10 to 15% existing between fission dosimeters and activation (fig.4),

The reconstitution of Charpy V test specimens from a broken test specimen half is done by welding two end pieces to the extremities, using a qualified process which has virtually no effect on the materials being assembled (fig. 6). The operation has been carried out by the BATTELLE Laboratories (USA).

The fluence received by the material at the location of the new notch has been calculated, enabling an experimental embrittlement curve of shell B to be plotted (RT_{NDT} - ϕ) as given in figure 7.

Moreover, the new fluence calculations have led to a correction in the interpretation of the surveillance program for other materials (shell C, HAZ, welded joint, reference metal). All these results are shown in fig. 8. They are modeled by a dependence of the type $\Delta RT_{NDT} = A \times \phi^n$, A and n depending on the material.

It can be seen that :

- the behaviour of the two shells is similar, embrittlement of shell C being slightly more than shell B for the fluences representative of the CHOOZ A vessel,
- the exponent n, modelling the behaviour of shells B and C is in the order of 0.5 or 0.6, which is appreciably much higher than in the case of similar materials irradiated at 290°C in reactor (n = 0.35 in the FIS formula).

For the fluence calculated at 20 years and 80% availability, the average behaviour of the vessel materials observed forecasts a maximum RT_{NDT} of 77°C, this value being obtained by shell C.

VESSEL INSPECTION

At the Administration's request, a vessel control and in particular one on the zone of the first thirty millimeters from the internal wall of the core zone, was carried out at right angles to a 300mm circumferential band around the B/C weld with the help of ultrasonic translators as well as a visual T.V. control of the cladding. This ultrasonic control was carried out in November and December 1987 after mechanical brushing of the cladding in order to eliminate deposits of interfering oxide generators :

- control of 1/3 of the parent metal thickness with the help of L0°, L24° and 2T45° translators,
- control of the first 30mm, from the internal vessel wall : with the help of OL60° and OL70° captors for controlling the under-cladding zone.

The method used guarantees the detection of any plan defect of a height at least equal to 3mm : the control carried out has not revealed any defects under the cladding in the controlled zone.

SYNTHESIS OF COMPLEMENTARY WORKS

These important works have enabled the Operator to set the RT_{NDT} of the vessel at :

- 64°C today,
- 77°C at the design end of life (20 years, i.e. 1992).

These RT_{NDT} values give adequate safety margins.

Taking into account the Operator's proposition to maintain the safety water injection temperature at 50°C, the margins were considered acceptable by the Safety Authorities.

CONCLUSIONS

Following the revelation in 1987 that the surveillance program results of the CHOOZ A vessel did not conform to predictions, and considering the uncertainties related to neutronic results, important complementary work was carried out by the Utility between June 1987 and March 1988 to assure that the plant could be restarted.

A complete re-evaluation of the neutronic properties of the vessel and the contrôl test specimens, on the basis of calculation and direct activity measurements, a direct determination of irradiation embrittlement of a shell of the core zone by reconstituted test specimens and an ultrasonic control of a significant part of the vessel wall, has shown that the safety margins of the CHOOZ A vessel with regard to fast fracture are adequate.

REFERENCES

1. 10 CFR 50 (§ 50.61) : Failure toughness requirements for protection against thermal shocks.

2. A. BEVILACQUA et al. Amélioration de la surveillance de la cuve d'un réacteur à eau pressurisée de la SENA : 5ème Symposium ASTM-EURATOM : 24-28 septembre 1984 - GEESTHACHT (RFA).

3. BRILLAUD C., HEDIN F., HOUSSIN B., A comparison between french surveillance program results and prediction of irradiation embrittlement. 13ème Symposium International ASTM sur les effets de l'irradiation sur les matériaux SEATTLE 1986.

		Transient	
	Defect	3rd Category	4th Category
Criterion	Envelope	C	D
	Exceptional	D	BD

Loading Conditions Category	Design fast fracture margins: K_I/K_{IC} ($T \le RT_{NDT} + 50°C$) RCC-M - Annexe ZG
Level A	0.4
Level C	0.5
Level D	0.3 or arrest ($K_{IC} < 0.8\ K_{1a}$)

TABLE I: Assessment of Margins Concerning Fast Fracture Resistance of the Vessel

	$RT_{NDT} - T_{IS}$			
Transients	75°C		35°C	
	Envelope Defect	Exceptional Defect	Envelope Defect	Exceptional Defect
2" Primary Break (3rd Category)	0.76 (0.5)	0.85 (0.8)	0.46 (0.5)	0.50 (0.8)
LOCA 4th Category	0.00 (0.8)	1.06 (1)	0.52 (0.8)	0.61 (1)

TABLE II: Margins in Relation to the Fast Fracture for the CHOOZ A Vessel

	C	Si	S	P	Mn	Ni	Cr	Mo	Cu	Co
Shell B (Acceptance)	0.150	0.28	0.008	0.015	1.17	0.55	0.15	0.40	0.08	0.02
Shell B (Test Specimen)	0.176	0.29	0.009	0.017	1.26	0.64	0.13	0.44	0.087	0.03
Shell C (Acceptance)	0.160	0.32	0.006	0.012	1.26	0.57	0.16	0.39	0.09	0.020
Shell C (Test Specimen)	0.169	0.31	0.005	0.010	1.27	0.65	0.15	0.39	0.09	0.02
Weld B/C (Specimen)	0.055	0.51	0.013	0.017	1.42	0.095	0.033	0.47	0.11	-

TABLE III: Chemical Composition of Core Zone CHOOZ A Vessel
(Acceptance and Test Specimen Analysis)

	RT_{NDT} (°C)
Shell B	- 18
Shell C	- 23
Weld B/C	- 20

TABLE IV: Core Zone CHOOZ A Vessel: Estimated
Initial RT_{NDT}

	1	2	3	4	5	6	7	8	9	10
High position	R - W	R - Z	BM	BM (1)	W	Z	BM	BM	BM - W	BM - Z
Low position	BM	BM	R - W	R - Z (2)	BM	BM	W	Z	BM - Z (2)	BM - W
Irradiation period	02/70 07/72	02/70 03/80	02/70 07/82	02/70 05/87	02/70 11/75	02/70 02/86			11/75 05/87	(3)

BM = Base metal shell C
W = Weld
Z = HAZ
R = Reference metal (A 302 grade B plate)

(1) Test specimens drawn but partly analyzed.
(2) Test specimens drawn but unalyzed.
(3) Unirradiated test specimens analyzed in 1982.

TABLE V: Surveillance Programme: Capsule Contents

	Ø 1 n.cm^{-2}.s^{-1} (E >1MeV)	Ø 0.1 n.cm^{-2}.s^{-1} (E >0.1MeV)	dpa.s^{-1}
Maximal Calculated Flux on Vessel	9.02 x 10^{10}	1.73 x 10^{11}	1.25 x 10^{-10}
Statistical (2σ) Incertitude (:)	3.2	5.6	3.2

TABLE VI: Neutronic Characteristics, CHOOZ A Vessel Assessment

Ø 1 n.cm^{-2}.s^{-1} (E >1MeV)	Measured	Calculated	Ø 1 Calculated / Ø 1 Measured
High test specimen	2.6 x 10^{11}	2.5 x 10^{11}	0.96
Low test specimen	1.15 x 10^{11}	1.17 x 10^{11}	1.02
Vessel (median plane)	8.0 x 10^{10}	8.81 x 10^{10}	1.10
Vessel-weld (B/C)	7.6 x 10^{10}		

TABLE VII: Comparison of Calculated and Measured Flux Values on Vessel and Test Specimens (Mn-54)

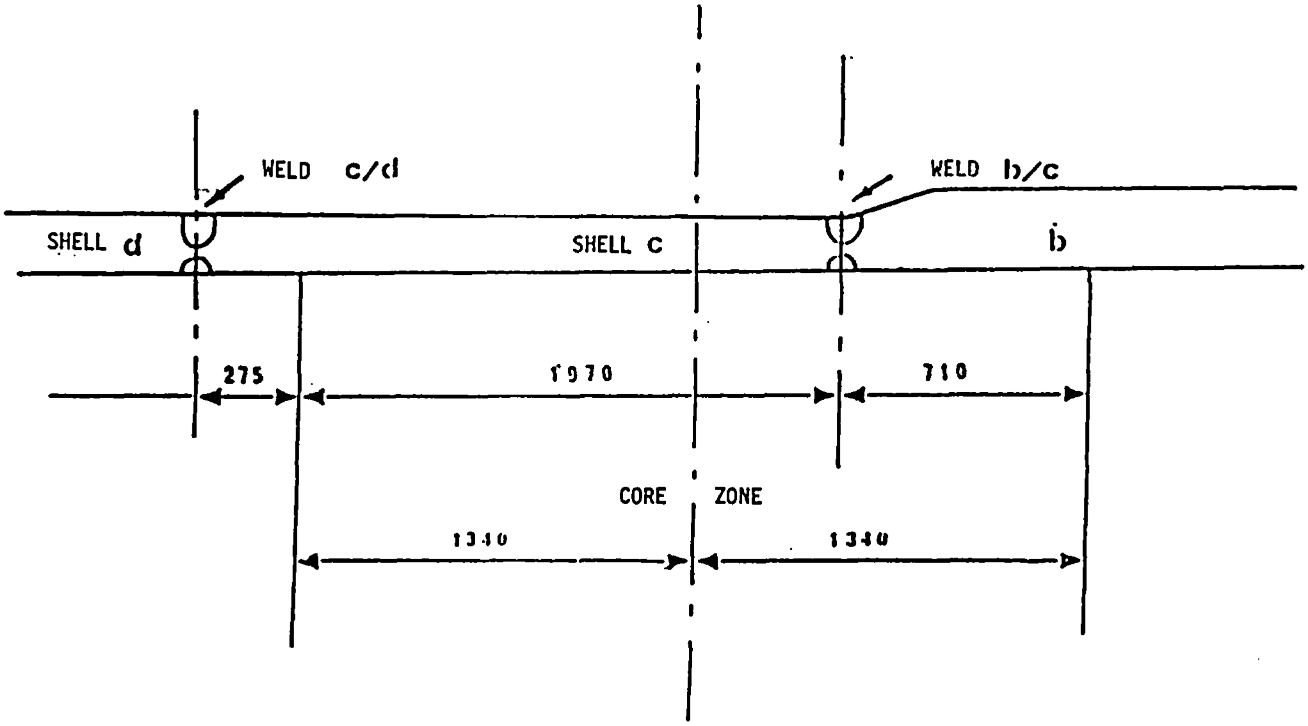


Figure 1: CHOOZ A Reactor: Core Zone Vessel

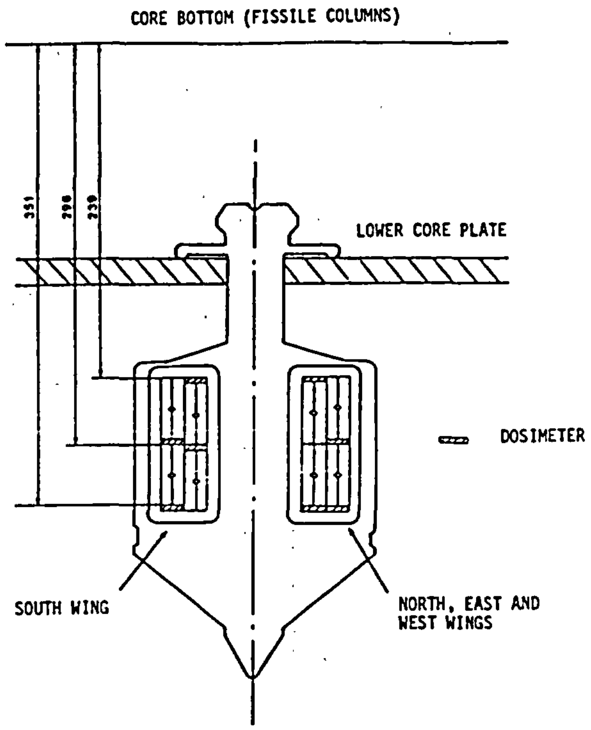
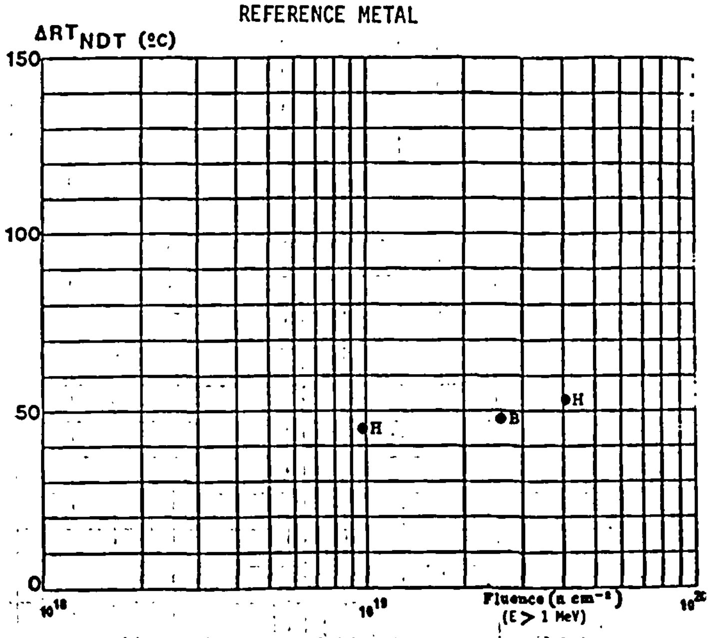


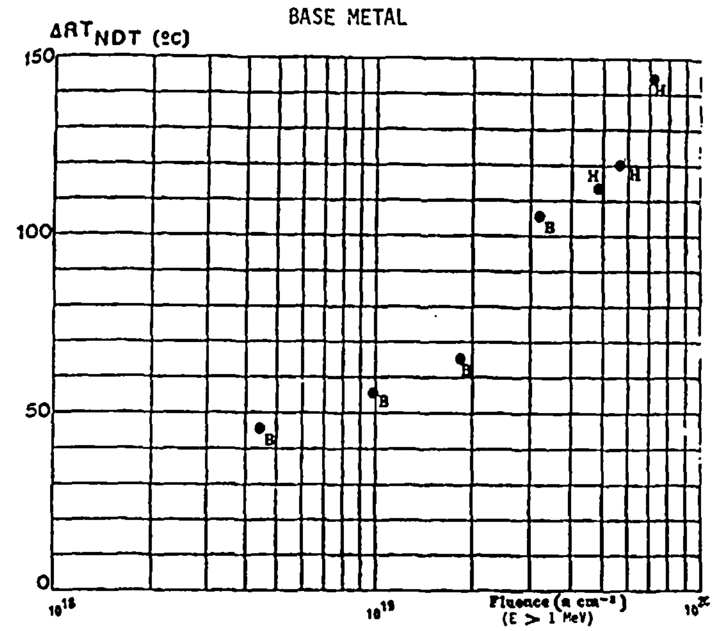
Figure 2: CHOOZ A: Surveillance Capsule Locations



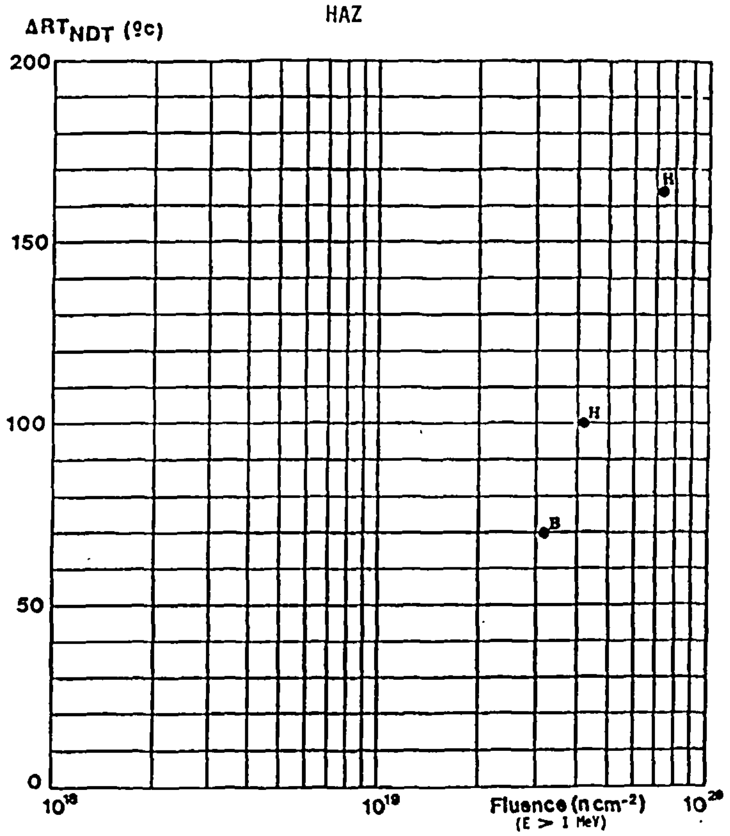
TEST SPECIMEN LOCATION IN THE CAPSULES:
H = HIGH, L = LOW

Figure 3: RT_{NDT} Shift from Test Specimens of Surveillance Program (1987)

Figure 3: Continued

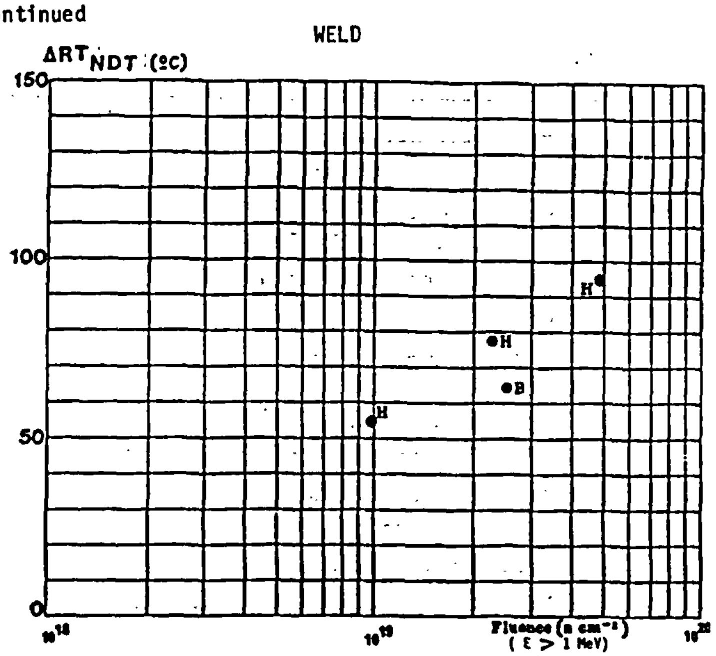


TEST SPECIMEN LOCATION IN THE CAPSULES:
H = HIGH, L = LOW



TEST SPECIMEN LOCATION IN THE CAPSULES:
H = HIGH, L = LOW

Figure 3: Continued



TEST SPECIMEN LOCATION IN THE CAPSULES:
H = HIGH, L = LOW

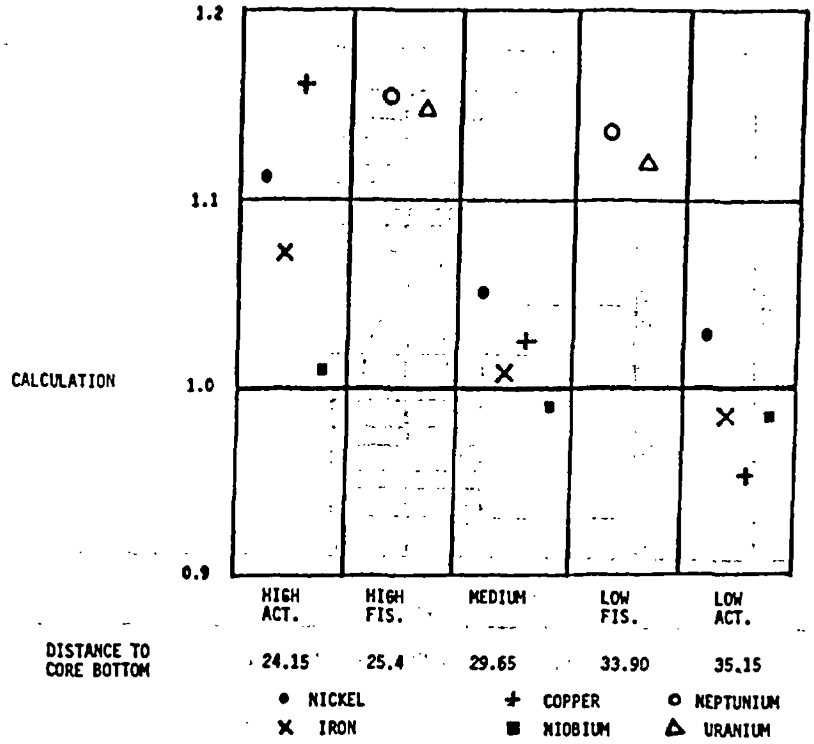
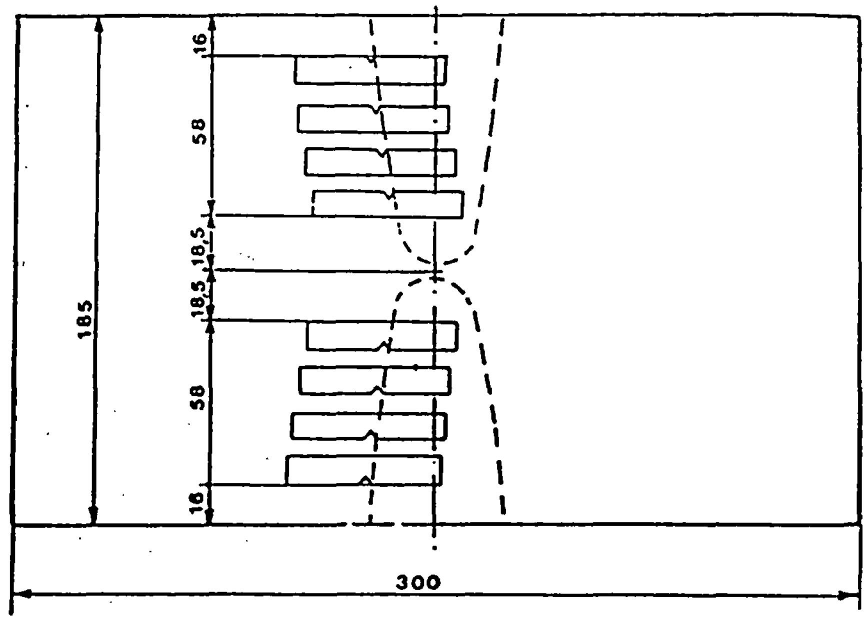
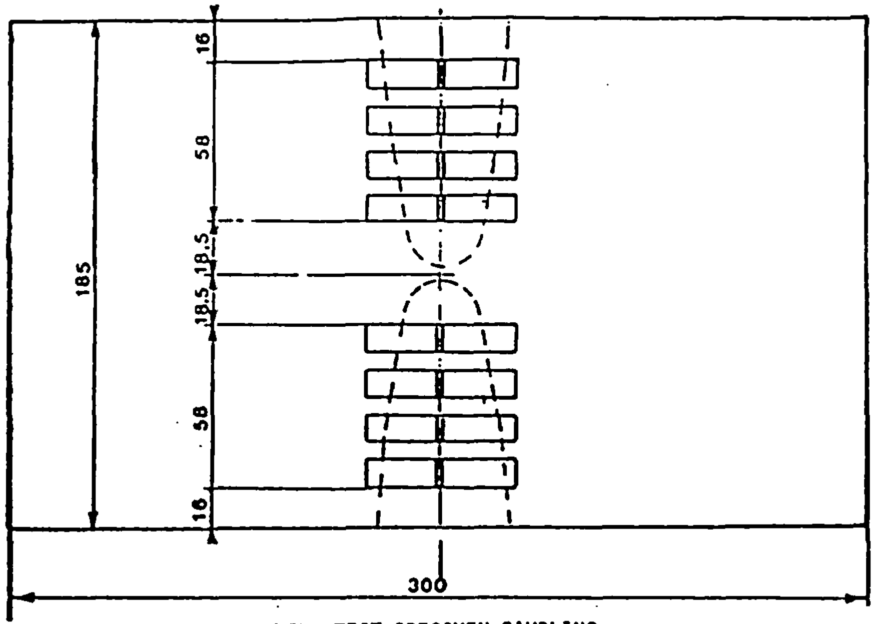


Figure 4: CHOOZ A Vessel: Comparison Between Calculated and Measured Fluxes (E > 1 MeV)



HAZ TEST SPECIMEN SAMPLING



WELD TEST SPECIMEN SAMPLING

Figure 5: HAZ And Weld Test Specimen Sampling

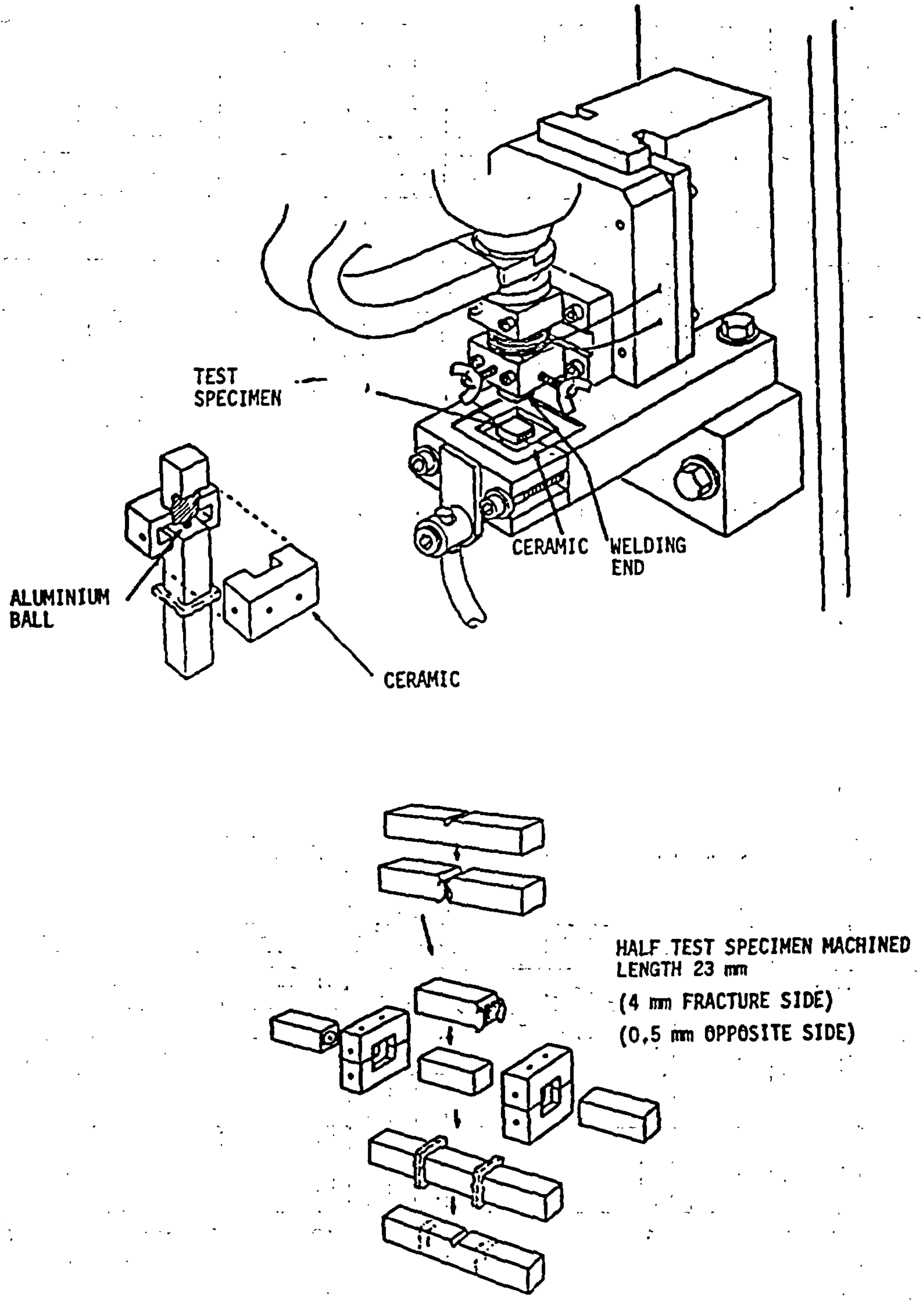


Figure 6: SHELL B: Test Specimen Reconstitution

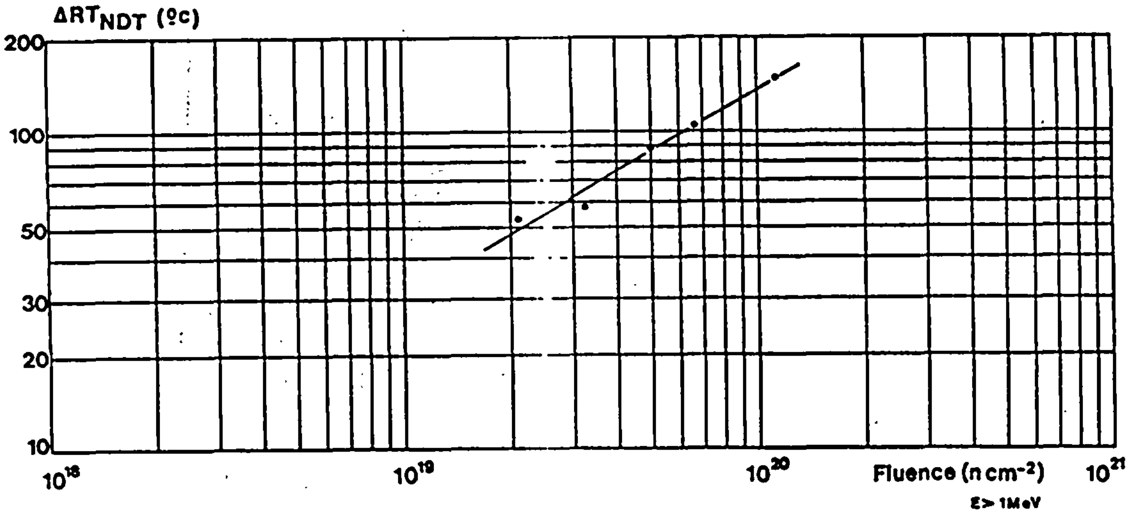
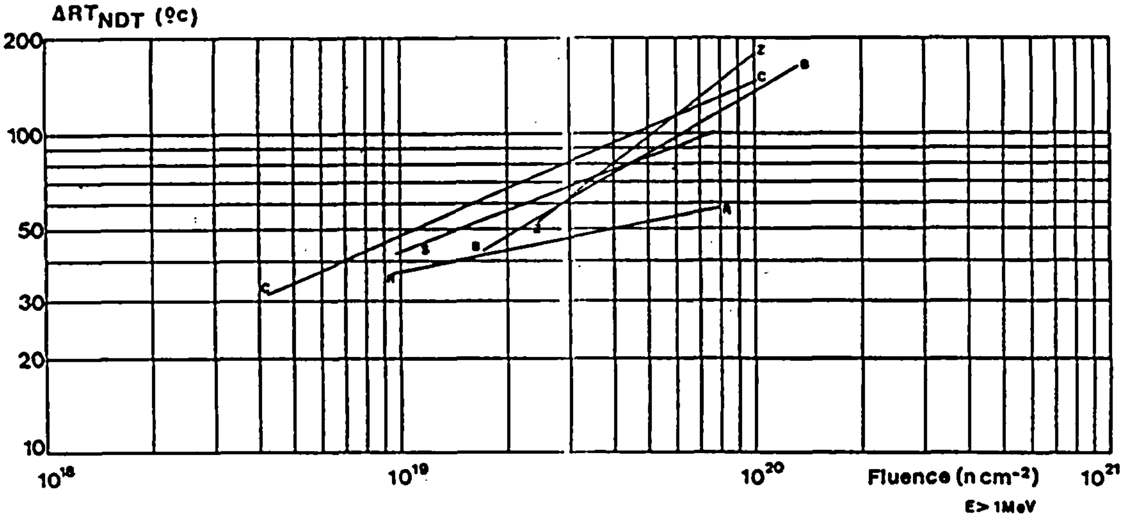


Figure 7: SHELL B: RT_{NDT} Shift from Charpy V Test Specimens Reconstituted



B : SHELL B
C : SHELL C
S : WELD
R : REFERENCE METAL
Z : HAZ

Figure 8: CHOOZ A Vessel Embrittlement

SENSITIVITY ANALYSIS APPROACH TO P-T LIMITS AND PRESSURIZED THERMAL TRANSIENT

Bogdan Glumac Mitja Najžer

Summary

Reactor pressure vessel steel undergoes significant changes in brittle to ductile transition temperature shift ΔRT_{NDT} when irradiated with fast neutrons leaking from the PWR core. This means that, according to ASME III-G, permissible pressure-temperature relations have to be recalculated periodically to help prevent the non-ductile failure of the RPV during pressure-temperature transients.

Heatup-cooldown limits calculations require intensive calculations of time dependent thermal and stress fields, RT_{NDT} determination, etc. and often with highly unreliable material properties data. The error induced by those unreliable data is then compensated for with introduction of additional safety factors.

By the use of sensitivity analysis approach one can, performing only moderately complicated calculations, not only calculate certain quantities, but also their variance-covariance matrices, based upon known (either measured or modelled) covariance matrix of input parameters (i.e. material data).

This contribution deals with an approach to first order (2) sensitivity analysis of the time dependent temperature profiles calculation ($\{T_i(t)\}$ and $< \delta T_i(t)|\delta T_j(t) >$) in the core belt of the pressure vessel wall and also briefly outlines what has to be done to calculate the stress profiles and linear elastic fracture mechanics (LEFM) stress intensity factors together with their variance-covariance information, and, on the other hand, determine adjusted RT_{NDT} and reference stress intensity factor K_{IR} together with their variances $< \delta RT_{NDT}|\delta RT_{NDT} >$ and $< \delta K_{IR}|\delta K_{IR} >$.

General Concept and Scope

When calculating the pressure-temperature limits or evaluating the consequences of a PTT (pressurised thermal transient) or low temperature overpressure (LTOP) event to the integrity of the reactor pressure vessel the methodology outlined in various sections of the ASME Boiler and Pressure Vessel Code is used. This methodology is based on the concepts of LEFM and uses often slightly simplified (i.e. adapted for the widespread use by the utilities engineering staff) mathematical tools. The error introduced by these simplified models, together with the uncertainties of the material constants used as input for these tools, is then compensated by conservativity factors in order to assure adequate margin of protection against the non-ductile failure. For instance: overall conservativity included in the calculational procedures for the pressure - temperature limits reaches almost a factor of ten.

In order to remove or at least better estimate whether such conservativity is in all cases justifiable we commenced the development of mathematical tools, still strictly based on the general concepts of the ASME code, which will eventually result in a set of computer codes that will, besides giving the desired results, also provide information about their uncertainty, contained in the covariance matrices of relevant quantities.

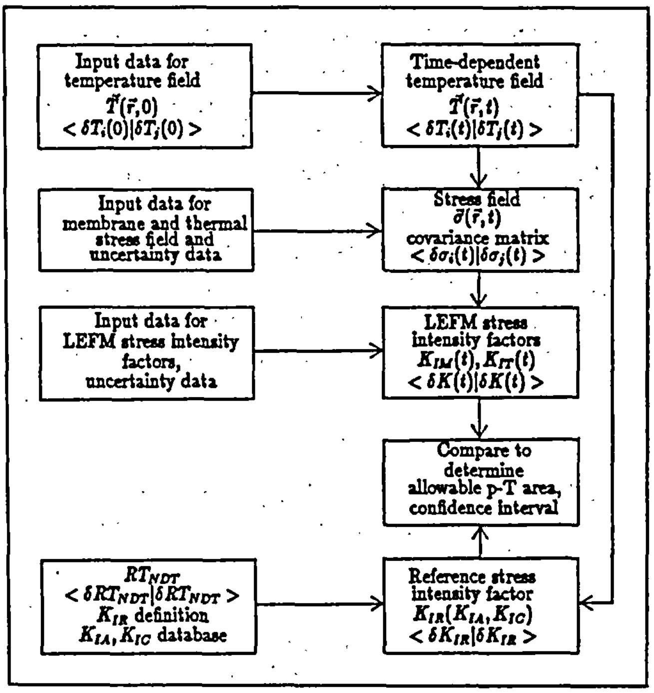


Figure 1: Schematics of the pressure - temperature limit curves calculation based on the sensitivity analysis approach

Mathematical apparatus needed to perform such calculations is contained within the scope of the probability theory, the theory of random variables and the sensitivity analysis (1,2,3,4) and is already being widely used in the areas of the neutron crossections evaluation, neutron dosimetry (neutron spectrum unfolding), burnup calculations, heat transfer problems of the nuclear fuel, etc.

On Figure 1 a diagram of a "classical" pressure - temperature limit curves calculation is sketched. The calculations necessary to evaluate the uncertainty information of the pressure - temperature curves, namely the propagation of the variance - covariance matrices through different segments of the calculation has also been included in order to point to the amount of the work needed to accomplish the task.

It can be seen that the calculation follows two paths. The first contains:

- temperature field: definition of the input parameter vector, modelling of its covariance matrix, calculation of the time dependent temperature profile through the pressure vessel wall, calculation of its time dependent covariance matrix;

- stress field: definition of the additional input parameters, modelling of the input parameters covariance matrix based on the covariance matrix of the temperature field and uncertainty information of additional parameters, calculation of the time dependent membrane and thermal stress fields and their covariance matrices;
- LEFM stress intensity factors: definition of additional input parameters (yield stress σ_y, which can be measured, crack geometry, etc.), calculation of the membrane (K_{IM}) and thermal (K_{IT}) stress intensity factors and their covariance matrices.

The second segment contains :

- ΔRT_{NDT}: determination by Charpy measurement, using sensitivity tools for data processing in order to determine also its variance;
- reference stress intensity factor K_{IR}: application of the sensitivity analysis tools to its definition database in order to obtain its variance. Covariance information about the temperature field is also taken into account (cracktip temperature).

The third step is the intercomparison defined by the ASME Code using the criterion :

$$C \cdot K_{IM} + K_{IT} \leq K_{IR} \tag{1}$$

where C is an appropriate safety factor. The final result is the allowable p-T area together with its confidence interval.

First Order Sensitivity Analysis

We prepare (1,2,3) an input parameter vector $\vec{p} \equiv \{p_1, \cdots, p_n\}$ which contains all the material properties data that take part in a certain calculation. To reach the desired final result $\vec{q} \equiv \{q_1, \cdots, q_n\}$, we perform a set of mathematical operations on the input parameter vector. Symbolically, we have:

$$\{q_1, \cdots, q_n\} = \vec{L}\{p_1, \cdots, p_n\} \tag{2}$$

or, by components, for a certain variable from the final state vector:

$$q_k = L_k(p_1, \cdots, p_n)\{p_1, \cdots, p_n\}. \tag{3}$$

If we treat the parameter vector as a joint random vector, we can assign to it its joint probability density function $f(p_1, \cdots, p_n)$ and define the expectation value of a certain random variable $< p_i >$. We can also define the second moment, known as the variance - covariance matrix:

$$< p_i >= \int_{-\infty}^{\infty} \cdots \int_{-\infty}^{\infty} p_i \cdot f(p_1, \cdots, p_n) \cdot dp_1 \cdots dp_n \tag{4}$$

and:

$$\begin{aligned} C(\vec{p})_{ij} &= < p_i - < p_i > | p_j - < p_j > > = < \delta p_i | \delta p_j > \\ &= < p_i \cdot p_j > - < p_i > \cdot < p_j > . \end{aligned} \tag{5}$$

Analog expressions are also valid for transforms contained in $\vec{q}$. By expanding a certain q_k from (3) in Taylor series [1] around its expectation [2]value (the value of "unperturbed" calculation) $< q_k >$ we obtain:

$$q_k \approx < q_k > + \sum_{j=1}^{n} \frac{\partial q_k}{\partial p_j}\Big|_{p_j = <p_j>} \cdot (p_j - < p_j >) \tag{6}$$

and, by some rearrangement, using the notation from equation (5), we end up with:

$$< \delta q_k | \delta q_l > = \sum_{j=1}^{n} \sum_{i=1}^{n} \frac{\partial q_k}{\partial p_j} \cdot \frac{\partial q_l}{\partial p_i} \cdot < \delta p_j | \delta p_i > \tag{7}$$

which gives us the relation governing the transformation of the covariance matrix of input parameters into the covariance matrix of output parameters.

The calculational procedure for the first order sensitivity analysis is now straightforward:

- the vector of input parameters $\vec{p}$ and its covariance matrix $< \delta p_i | \delta p_j >$, based upon the information about the uncertainty of input data is prepared;
- mathematical formulation $\vec{L}$, which brings us from input data to desired final result is defined and the calculation is performed;
- the covariance matrix of the final state is determined according to equation (7).

The easiest way to determine the derivatives of the type $\frac{\partial q_k}{\partial p_j}$ from equation (7) is by numerical approach. We know that :

$$\frac{\partial q_k}{\partial p_j} \approx \frac{q_k(p_1, \cdots, p_j + dp_j, \cdots, p_n) - q_k(p_1, \cdots, p_j, \cdots, p_n)}{dp_j} \tag{8}$$

so, by simply perturbing the set of input parameters (one at a time) and performing for each final parameter q_k one calculation on unperturbed input parameter set $\{p_1, \cdots, p_n\}$ and n calculations on n perturbed sets beginning with $\{p_1 + dp_1, \cdots, p_n\}$ and ending with $\{p_1, \cdots, p_n + dp_n\}$, we define the sensitivity matrix from equation (8).

---

[1] First order sensitivity analysis means that we cut the Taylor series after the first term (2)

[2] Assuming $f(p_1, \cdots, p_n)$ is joint normal. Assumption is feasible for all parameters which take part in this calculation

Sensitivity Analysis of a Time Dependent Temperature Field in the Reactor Pressure Vessel Wall

At first we choose a certain numerical model to describe the time dependent temperature field in the reactor pressure vessel wall during a transient. For simplicity we took one dimensional model (radial dependence only) in finite differences approximation (long hollow cylinder). We took convective heat transfer boundary condition for the inside surface and ideal insulation boundary condition for the outside surface. After all the equations that govern the time dependent heat transfer are written down for M equidistant intervals, we have the following set of input parameters:

- radii to mesh points $r_1, \cdots, r_{M+1}$;
- mesh step Δr;
- bulk coolant temperature $T_f(t)$;
- convective heat transfer coefficient $h(T_f(t))$;
- temperature field $T_1(t), \cdots, T_{M+1}(t)$;
- thermal conductivities $k(T_1(t)), \cdots, k(T_{M+1}(t))$;
- density ρ;
- specific heat c;
- gamma heating power densities $\gamma_1, \cdots, \gamma_{M+1}$.

For M mesh points we have to deal with $4 \cdot M + 9$ parameters: some of them change with time, some do not.

Now we have to model the initial covariance matrix of the input parameters. When modelling the covariance matrix one has to consider all possible correlations so, by our opinion, this is the most demanding part of the entire sensitivity calculation. Additional obstacle that one encounters is almost total absence of uncertainty data on measured parameters like convective heat transfer coefficient, thermal conductivities, etc.

To demonstrate the amount of work necessary to model the correlations we will briefly sketch how the part of the input parameters covariance matrix that describes convective heat transfer is being modelled.

Bulk coolant temperature $T_f(t)$ is measured by a set of thermocouples with a certain precision declared by the manufacturer. This value was taken for bulk coolant temperature absolute error $\sqrt{<\delta T_f(t)|\delta T_f(t)>}$. Heat transfer coefficient was given in tabular form $\{\alpha_i, \beta_i\}$, where α_i is coolant temperature and β_i is heat transfer coefficient value at this temperature. Absolute errors $\delta\alpha_i$ and $\delta\beta_i$ were also given. Tabular values showed clear linear dependence, so a linear least squares fit was performed on this data set to yield the result $h(T_f(t)) = a_0 + a_1 \cdot T_f$. From this equation, using the formalism from Equation (7), it is evident that the contribution of the heat transfer coefficient $h(T_f(t))$ and the bulk coolant temperature $T_f(t)$ to the input parameters covariance matrix is given by:

$$\begin{vmatrix} <\delta h|\delta h> & <\delta h|\delta T_f> \\ <\delta T_f|\delta h> & <\delta T_f|\delta T_f> \end{vmatrix}$$

where:

$$\begin{aligned} <\delta h|\delta h> &= <\delta a_0|\delta a_0> + 2 \cdot T_f \cdot <\delta a_0|\delta a_1> \\ &+ a_1^2 \cdot <\delta T_f|\delta T_f> + T_f^2 \cdot <\delta a_1|\delta a_1> \end{aligned} \tag{9}$$

and:

$$<\delta T_f|\delta h> \equiv <\delta h|\delta T_f> = a_1 \cdot <\delta T_f|\delta T_f> . \tag{10}$$

To obtain the relations in equations (9) and (10) we had to develop a least squares tool that propagates the covariance matrix of the tabular values $\{\alpha_i, \beta_i\}$ for the coolant temperature and the heat transfer coefficients:

$$\begin{vmatrix} <\delta\alpha_i|\delta\alpha_j> & 0 \\ 0 & <\delta\beta_k|\delta\beta_l> \end{vmatrix}$$

into the covariance matrix of the linear least squares fit:

$$\begin{vmatrix} <\delta a_0|\delta a_0> & <\delta a_0|\delta a_1> & 0 \\ <\delta a_1|\delta a_0> & <\delta a_1|\delta a_1> & 0 \\ 0 & 0 & <\delta T_f|\delta T_f> \end{vmatrix}$$

in order to be able to obtain the convective heat transfer segment of the input parameters covariance matrix.

Similar procedure is used to prepare other segments of the input parameters covariance matrix. Tools for least squares parabolic fit and exponential fit were also prepared using the same technique and assumptions as for linear fit.

In Figure 2 the correlation coefficients surface for the input parameters vector for the time dependent temperature field calculation is given for illustration:

Sensitivity analysis of the time dependent temperature field follows the path sketched in Figure 3 and using the newly developed computer codes "LINCOV", "PARCOV", "EXPCOV", "PARVEC" and "COVART/1".

By the use of the above mentioned computer codes we have calculated through-wall temperature profiles for various operating states or transients that may occur in a nuclear steam supply system and that have, as a consequence, introduction of a non-uniform temperature profile in the pressure vessel wall which then results in additional tensile stresses.

All calculations were performed using the data set of the Krško Nuclear Power plant (Westinghouse two loop PWR with ≈ $630MWe$). On Figure 4 the equilibrium temperature profile due to "worst case[3]" gamma heating for Krško NPP is given :

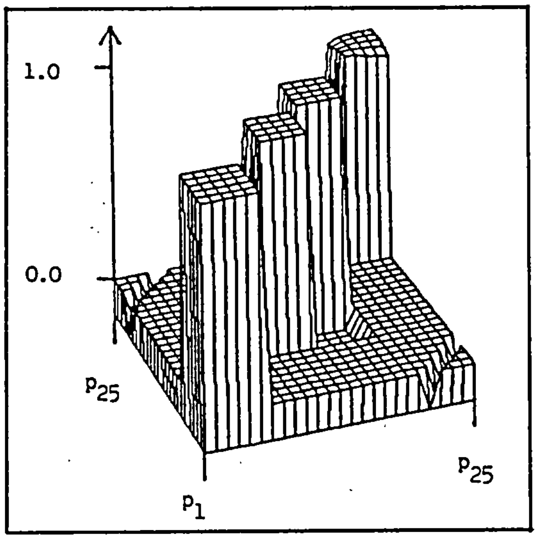


Figure 2: Correlation coefficients surface of the input parameter vector for time dependent temperature profile calculation through the RPV wall (case $M = 4$ where $p_1 = r_1$ and $p_{25} = \gamma_5$, Krško NPP data)

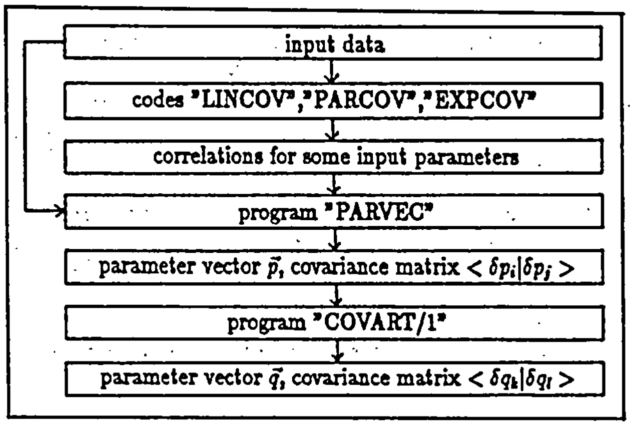


Figure 3: Schematics of the time dependent temperature field and its covariance matrix calculation

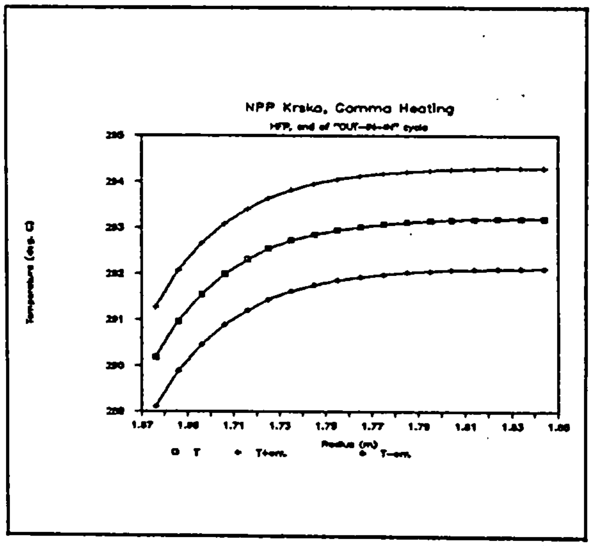


Figure 4: Equilibrium temperature profile due to gamma heating at rated power, Krško NPP, end-of-life for 4th core cycle, "OUT-IN-IN" core load

On Figure 5 the temperature segment of the final parameter vector covariance matrix (in a form of correlation coefficients matrix) for the above mentioned gamma heating case is given.

The second example is our evaluation[4] of the pressurised thermal transient that occured in Rancho Seco in 1985 due to loss of DC power to the integrated control system. On Figure 6 we give the through-wall temperature profile at the time of repressurisation maximum (some 1000 sec into the transient), on Figure 7 again the temperature segment of the output parameter vector correlation coefficients matrix for this profile is given.

Conclusions

A project that is now underway and that has a scope to provide first order sensitivity analysis tools for pressure - temperature transients evaluation (according to relevant US legislation) in the core belt of a pressurised water reactor has been outlined.

Some results obtained with the part of a project that is close to completion, namely the sensitivity tool for the time dependent temperature field (in the core belt of the pressure vessel) calculation, have also been presented together with tools for estimation of the uncertainties hidden in input parameters for such calculations.

---

[3] This means the highest possible gamma heating power densities for Krško NPP which occur at the end of equilibrium "OUT-IN-IN" core load.

[4] The calculation was performed as if this transient would happen in Krško NPP - this means that the RCS T_{avg} had to be renormalised to NPP Krško setpoints.

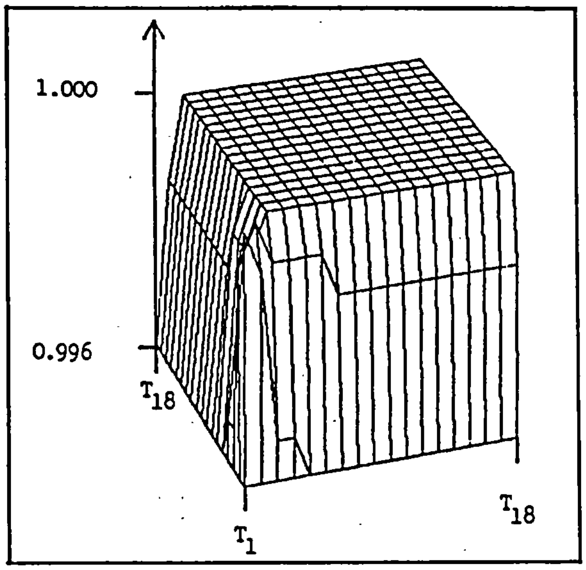


Figure 5: Temperature segment of the output parameter vector correlation coefficients matrix, stationary through-wall profile due to gamma heating, hot full power, end of 4th fuel cycle, Krško NPP

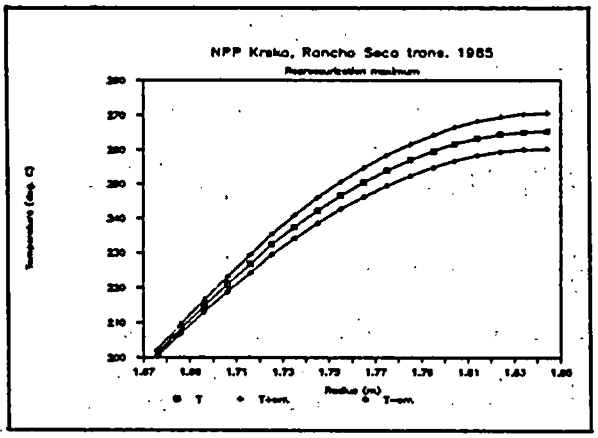


Figure 6: Through-wall temperature profile at t=1080 sec for Rancho Seco (*Dec.26th*, 1985) type pressurized thermal transient, NPP Krško data set

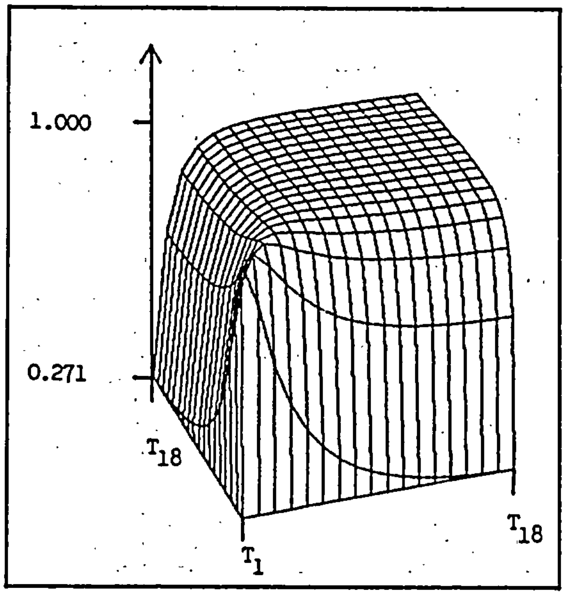


Figure 7: Temperature segment of the output parameter vector correlation coefficients matrix, 1985 Rancho Seco pressurized thermal transient, t=1080 sec, NPP Krško data set

References

1. Athanasios Papoulis: Probability, Random Variables and Stochastic Processes, McGraw - Hill, 1985, ISBN 0 - 07 - 466465 - X.
2. Yigal Ronen, Derivation of the n^{th} Order Uncertainty Analysis for Normally Distributed Parameters, Nucl. Sci. Eng., Tech. Note, 314 (1984).
3. D. G. Cacuci et al.: Sensitivity Theory for General Systems of Nonlinear Equations, Nucl. Sci. Eng., 75, 88 (1980).
4. NUREG-1195: Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26th, 1985.

REACTOR VESSEL EMBRITTLEMENT MANAGEMENT FOR LIFE ATTAINMENT AND EXTENSION

B. C. Rudell
S. T. Byrne

ABSTRACT

A multifaceted approach has been initiated by Baltimore Gas & Electric to address reactor vessel embrittlement issues at Calvert Cliffs. Augmented reactor vessel surveillance capsules, excore neutron activation monitors, enhanced state-of-the-art nondestructive examination, alternative core physics planning, thermal hydraulic analysis and fracture mechanics methodologies have been integrated to manage vessel embrittlement over the current licensed life and beyond.

Long term exposure to neutron irradiation causes a decrease in fracture toughness of reactor pressure vessel steels. Reactor Pressure Vessels (RPVs) are designed to accommodate a certain amount of toughness loss by establishing controls on operation and by the use of safety systems. RPV fracture toughness safety margins are determined by analyses and also by measurements obtained from surveillance programs and inservice inspection.

Both the regulatory environment and industry knowledge pertaining to radiation damage have changed substantially since early RPV constructions. The Calvert Cliffs RPVs are among those which have been significantly impacted by these changes. This has necessitated major actions to maintain or recover the RPV fracture toughness safety margin.

Actions taken on the materials analysis approach and the non-destructive examination approach and their applicability to general plant aging evaluations are discussed in this paper.

INTRODUCTION

Long term exposure to neutron irradiation causes a decrease in fracture toughness of the Reactor Pressure Vessel (RPV) materials adjacent to the nuclear core. This embrittlement process has been recognized since the beginning of commercial nuclear power and vessels were designed to accommodate the resultant toughness losses. For example, the materials of construction had to meet minimum initial toughness requirements and vessel operating limits were adjusted periodically to account for embrittlement. Furthermore, reactor vessel surveillance and Inservice Inspection (ISI) programs were established to monitor changes in fracture toughness and to verify the absence of flaws which could challenge the integrity of the vessel in the unlikely event of a severe transient.

Both the regulatory environment and industry knowledge pertaining to radiation damage have changed substantially since the start of commercial nuclear power. Reactor Pressure Vessels built during the 1960's were significantly impacted by these changes and this necessitated major actions to maintain or recover reactor vessel safety margins influenced by radiation induced changes in fracture toughness. The following describes what actions have been taken by Baltimore Gas & Electric (BG&E) Company to manage reactor vessel embrittlement at the Calvert Cliffs Nuclear Power Plant in order to meet plant life attainment and life extension goals.

EMBRITTLEMENT MANAGEMENT PROGRAM OVERVIEW

BG&E adopted a multifaceted approach to address RPV embrittlement issues at Calvert Cliffs and thus ensure safe operation over plant life, including planning for extended licensing. Major elements of the program include:

- fuel management to reduce neutron flux to the RPV;
- documentation of RPV beltline materials properties by record searching;
- acquisition of additional vessel materials for irradiation surveillance;
- implementation of a supplemental irradiation dosimetry monitoring program;
- development and application of advanced Non-Destructive Examination (NDE) equipment for ISI;
- method development for applying ISI results to fracture mechanics analyses; and
- application of advanced fracture mechanics analysis techniques.

The approach being taken is designed to satisfy the Pressurized Thermal Shock (PTS) Rule[1] (10CFR50.61) and to attain sufficient operating flexibility to achieve the life attainment/license extension goal for Calvert Cliffs. The first element of the BG&E approach, fuel management, is designed to directly reduce embrittlement whereas the balance are designed to demonstrate additional margin by obtaining better data as input to radiation damage prediction techniques (e.g., the PTS rule or Regulatory Guide 1.99[2]) or to vessel integrity analyses (e.g., 10CFR50 Appendix G[3] or Regulatory Guide 1.154[4]). The various elements of the embrittlement program are described in two distinct parts; the Materials Analysis approach and the NDE approach.

MATERIALS ANALYSIS APPROACH TO EMBRITTLEMENT MANAGEMENT

A detailed evaluation was performed of the Calvert Cliffs Units 1 and 2 RPVs to identify where better materials and neutron fluence data would be beneficial in establishing additional margin for normal operation and for postulated transients. Because the two vessels were designed and fabricated during the 1960's, the evaluation addressed the impact of a large number of changes in regulations and required seeking various critical materials data. Furthermore, in order to accommodate long term needs, the impact of anticipated regulatory changes (e.g., revision of the PTS rule) and surveillance monitoring over an extended lifetime were also addressed. The majority of the effort was concentrated on the Unit 1 RPV because of its greater predicted sensitivity to neutron radiation.

The controlling material (i.e., that with the greatest predicted adjusted reference temperature using Reference 2) in the Unit 1 RPV is weld seam 2-203 A/C, the intermediate shell longitudinal seam welds. A different material was included in the surveillance program, and consistent with the practice of the time, only a limited amount of data on chemical content and mechanical properties were obtained on weld materials other than the surveillance material. It was determined that the Calvert Cliffs Unit 1 controlling material (identical weld wire heats, type of flux and weld process) was used in the McGuire Unit 1 surveillance program. This was confirmed by examination of original fabrication records. This equivalence established a source for both initial properties (chemical content and RTNDT), and post-irradiation properties (RTNDT shift, upper shelf energy decrease and tensile strength changes) through the McGuire Unit 1 reactor vessel surveillance program. It also provided a source (through the cooperation of Duke Power) of a limited amount of material from the McGuire archive weld block for irradiation in the Calvert Cliffs reactor vessel. By obtaining data from the same material in both irradiation environments the effect of any differences between the two surveillance capsule environments could be quantified and a greater amount of irradiation data would be available to both McGuire and Calvert Cliffs.

An augmented materials surveillance program has been implemented between the McGuire and Calvert Cliffs RPVs through weld Charpy specimen irradiation. The surveillance programs for both units are being integrated to produce data useful to each vessel. In the case of the McGuire post-irradiation surveillance data, a supplemental evaluation will be performed to assure that the irradiation environment is similar to that in Calvert Cliffs. In parallel, two sets of McGuire surveillance weld specimens are being irradiated in Calvert Cliffs; the irradiation exposure will overlap the exposures for the first two McGuire capsules.

There are two unique features of the replacement capsule effort at Calvert Cliffs. First, a "piggy-back" approach is being used within one full surveillance capsule; at the completion of the first irradiation period half of the capsule will be reconstituted and reinstalled for an additional irradiation period. Second, a surrogate set of Charpy test specimens has been included with the specimens from the McGuire weld; the second material is nearly identical to the McGuire weld and is available in a much greater quantity. The concept is to provide data to demonstrate that a surrogate material can be used, if needed, in the future for continued embrittlement monitoring and for future vessel annealing surveillance.

Material properties represent part of the materials analysis embrittlement equation. The other major part is neutron fluence. BG&E has introduced new fuel management techniques to reduce the neutron flux to the RPV and thus slow embrittlement. Methods to further reduce the neutron flux are being evaluated. In order to accurately measure the effect of fuel management changes on the neutron flux, a supplemental dosimetry program was implemented. The program consists of neutron flux wire monitors positioned at numerous locations in the reactor vessel cavity and a pair of capsules each containing three sets of nine individual flux monitors. Both capsules are at the same azimuthal location, one inside the vessel and the other outside in the cavity to establish a correlation between the two neutron environments. The full set of dosimetry (as described) is being used to monitor an eighteen month fuel cycle and the current twenty-four month cycle. Subsequent cycles will be monitored periodically using cavity monitoring only, relying on the correlation established with the in-vessel dosimeter capsules.

The primary objective of the supplemental RPV dosimetry program is to produce accurate information on the magnitude of the neutron flux and its spatial distribution. For vessel embrittlement analysis, reducing the uncertainty in fluence calculations will reduce the overall uncertainty of the analysis. Any credit claimed for fuel management or azimuthal variations resulting in fluence reductions at critical locations can be supported by the supplemental dosimetry measurements. Furthermore, detailed measurements will facilitate planning insertion and withdrawal schedules for replacement capsules and for comparing the radiation environment in Calvert Cliffs to that in McGuire with regard to sharing capsule test results.

The combination of supplemental materials and dosimetry measurements is being generated specifically to:

- optimize operating flexibility using Regulatory Position 2.1 of Regulatory Guide 1.99;
- generate information to respond to an anticipated revision to the PTS rule based on Regulatory Guide 1.99; and
- permit a favorable result from a vessel integrity analysis based on Regulatory Guide 1.154 if required.

Regulatory Guide 1.154 provides an opportunity for improved technical input in two other areas. First, the PTS screening criterion is based on initiation of a crack, whereas analysis in accordance with Regulatory Guide 1.154 allows one to demonstrate that an initiated crack will be arrested before it penetrates the vessel wall. Recent developments in fracture mechanics analysis techniques[5,6] have provided a useful tool for performing crack arrest analysis and a crack arrest algorithm is being developed to apply this new technology.

A second area where technology improvements can improve PTS analyses involves the application of NDE results; this approach is discussed next.

NON-DESTRUCTIVE EXAMINATION APPROACH TO EMBRITTLEMENT MANAGEMENT

Evaluation of ultrasonic inspection results from RPV weldments has generally revealed a significant level of uncertainty in the size, orientation and connectivity of indications. This uncertainty has necessitated a degree of over-conservatism in regulatory codes and development of statistical distributions of flaw size.

Efforts have been undertaken to improve the reliability and repeatability of ultrasonic inspections. The efforts have led to development of NDE systems that have demonstrated superior ability to detect and size underclad cracks and weld defects. BG&E intends to carry these efforts one step further by incorporating these recently developed NDE advancements into the structural integrity analysis of the Calvert Cliffs vessels.

Preservice and periodic inservice NDE are mandated by law and are performed in accordance with rules of Section XI[7] of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Additional guidance for performing examinations is contained in the NRC Regulatory Guide 1.150,[8] "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

Although acceptable to Code and regulatory procedural requirements, conventional examination procedures may not meet the needs of providing an adequate technical basis for plant license extension arguments and responding to concerns for nuclear plant safety. Recently developed examination technology and methods of validating examination effectiveness are now available to meet these extended needs.

The RPV maintains a unique place among the nuclear steam supply system components because its failure is unacceptable. The continued demonstration of incredibility of vessel failure is a requirement of nuclear plant design and operation. Accurate detection and characterization of vessel flaws are essential to this demonstration. National and international cooperative programs have shown that examinations performed to current minimum Code requirements are inadequate to detect many of the defects of interest, particularly underclad cracks. In response to this problem, the NRC issued Regulatory Guide 1.150. After the Regulatory Guide was issued, several plants experienced incidents in which the emergency core cooling system injected cold water into the RPV while the vessel was still under substantial pressure. This Pressurized Thermal Shock (PTS) can cause thermal and pressure tensile stresses near the inside surface of the vessel which are non-negligible. This possible combination of embrittlement, operating transient, and flaws has provided a very strong incentive for the development of an improved RPV inspection capability.

Current rules for predicting the failure probability over the service life of the vessel specify a flaw distribution that is most probably overly conservative in specific cases. Accurate data on the actual flaw distribution must be a component of a reliable evaluation of the integrity of a vessel. To this end, it was important to inspect to the highest possible standard in order to increase the probability of detecting even very small discontinuities.

Pacific Gas & Electric, recognizing the potential of applying sophisticated target acquisition technology to ultrasonic inspection, placed a contract with Dynacon Systems, Inc. to develop, test, and supply an Ultrasonic Data Recording and Processing System (UDRPS).[9] This system was designed to be optimally suitable for examination of RPVs. In early 1983, funding for the project was continued by the Electric Power Research Institute (EPRI) so that

* development of the display and data evaluation software could continue; and

* the system capability for detection of under-clad cracks, a concern in a PTS scenario, could be demonstrated.

This work was followed by a much more complete evaluation by the EPRI NDE Center. This evaluation covered many aspects of defect detection in both the under-clad region of RPVs and the welds. Although defect detection was the primary aim of the evaluation, the ability of the system to size indications from the detection data was also evaluated.

The results of an NDE Center evaluation of UDRPS were published in January 1986.[10] Concurrently, BG&E had contracted with Dynacon for the design and fabrication of a versatile multi-channel UDRPS with the intention of using the system during the first 10-year ISI at Calvert Cliffs (Units 1 and 2) to provide new baseline data for use in future license extension arguments.

In preparation for the use of UDRPS for a Reactor Pressure Vessel examination, qualification testing of BG&E's system was performed at the NDE Center in May 1986. Interface work with the Code Inspection vendor, Southwest Research Institute (SwRI) was conducted in September 1986. It was of particular importance that the additional UDRPS examination should have minimal impact on the legally required ASME Section XI examination and on the inspection schedule. In cooperation with SwRI and Dynacon, an approach was devised such that the BG&E UDRPS could acquire data from the SwRI transducers in parallel with the equipment used for recording the Code examination.

Throughout the whole period of purchase of the UDRPS and planning for its use at the Calvert Cliffs 10-year ISI, staff at the NRC were kept informed and involved. This process started with an information presentation at NRC Region 1 where the results of the EPRI evaluation were presented and discussed. Early plans for applying the UDRPS system at Calvert Cliffs were also presented. Later in the project hands-on sessions were held at the BG&E laboratories at Fort Smallwood to allow the NRC staff to become more familiar with the system and the graphics software. Finally a demonstration of the combined UDRPS SwRI system was presented to the NRC and the Authorized Nuclear Inspection Agency at the SwRI facility in San Antonio, Texas.

The field examination of Calvert Cliffs Units 1 and 2 took place in November 1986 and April 1987 respectively. Virtually the entire length of RPV beltline weld was examined with the exception of only few limitations caused by the presence of the surveillance capsules used for monitoring of material properties. Some of the results of this analysis were presented at the Post-SMirt Seminar.[11] Each RPV inspection at Calvert Cliffs resulted in approximately 5 Gigabytes of ultrasonic data stored on optical disc. Due to the relative ease of transferring large amounts of data it was possible for the review and analysis of the data to proceed in parallel with data acquisition.

With the short term objective of applying UDRPS to a full scale ISI of the Calvert Cliffs RPV met, work has now begun to apply the results to probabilistic analysis of RPV integrity. A scoping study is currently underway to address application of the UDRPS exam results to a PTS fracture mechanics analysis. Once completed, guidelines can be established for reporting results in a manner that will be most useful in establishing specific PTS flaw distribution for each RPV. Parallel with these efforts continued refinement of UDRPS examination capabilities will occur through data characterization using real components and flaw qualification blocks. Once the probability distribution of real flaw size in a vessel can be determined from indicated flaw sizes it would be a straightforward matter to determine actual size distributions for use in plant specific analysis and remove fracture mechanics uncertainties.

Analogous to these efforts, results of a UDRPS examination of a generator field were evaluated by Failure Analysis Associates. This evaluation cited the capabilities and limitations of a UDRPS NDE to interface with a probabilistic stress/fracture mechanics analysis code, the EPRI-developed SAFER code.[12]

CONCLUSIONS

1) Life attainment and extension is feasible given a multifaceted approach to embrittlement management.

2) Timely planning is critical to augment an irradiation surveillance program.

3) Integrated programs with other RPVs provide a valuable resource for information and key materials.

4) Supplemented surveillance, advanced fracture mechanics analysis methods, and enhanced NDE techniques significantly increase the potential for demonstration of vessel integrity.

5) Guidelines are needed for integrated surveillance programs involving non-identical RPVs, for applying surveillance data when the controlling vessel material is not included in the surveillance capsules, for obtaining supplemental neutron flux measurements via ex-vessel dosimetry, and for adapting surveillance programs to extended service life.

6) Effort is needed to better characterize sophisticated NDE processes to demonstrate the quality of the data and to establish guidelines on how NDE data can be used in vessel integrity analyses.

REFERENCES

1. Title 10, Code of Federal Regulations, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," dated July 23, 1985.

2. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

3. Title 10, Code of Federal Regulations, Part 50, "Fracture Toughness Requirements," dated May 27, 1983.

4. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Report for Pressurized Water Reactors," dated January 1987.

5. R. J. Fabi, et. al., Combustion Engineering, Inc., "A Simplified Method for Calculating the Stress Intensity Factor at Crack Arrest," SMIRT-9 Conference, August 1987.

6. R. J. Fabi, et. al., Combustion Engineering, Inc., "A New Procedure for Evaluating Crack Arrest in Reactor Vessels," SMIRT-9 Conference, August 1987.

7. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code: Section XI Rules for Inservice Inspection of Nuclear Power Plants," New York, 1983.

8. Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," dated February 1983.

9. M. J. Moore, F. J. Dodd, "Real-Time Signal Processing in an Ultrasonic Imaging System," Materials Evaluation, 40, 976-81, 1982.

10. "Evaluation of the Ultrasonic Data Recording and Processing System (UDRPS)," Electric Power Research Institute, NP-4397, January 1986.

11. S. R. Buxbaum, R. B. Pond, Jr., A. J. Willets, "Application of an Ultrasonic Data Recording and Processing System to Reactor Pressure Vessel Examination," Non-Destructive Examination in Relation to Structural Integrity: Post-SMIRT Seminar No. 3 (to be published 1988).

12. "Life Assessment Methodology for Turbogenerator Rotors," Electric Power Research Institute, EPRI CS/EL-5593, March 1988.

ASSESSMENT OF FUTURE OPERATING STRATEGIES FOR NUCLEAR STEAM GENERATORS USING DETAILED DEGRADATION MODELS

Dr. Frank J. Berte'
Combustion Engineering, Inc.

Abstract

Combustion Engineering (C-E), working with five major utilities, has developed a unique Strategy Assessment Model (SAM™) for both C-E and Westinghouse (W) designed nuclear steam generators. The computer-based model, for which a patent has been obtained, incorporates both "macroscopic" and "microscopic" analytic components to characterize selected degradation processes. The macroscopic model encompasses statistical characterization of prior non-destructive evaluation (NDE) data on the generator. The microscopic model addresses the effects of bulk chemistry, thermal-hydraulics, and metallurgy on the localized environment where degradation takes place. For C-E steam generators, pitting and wastage have been modeled; intergranular attack (IGA), stress-corrosion cracking (IGSCC), and denting models are complete. For W steam generators, the microscopic component of the strategic model for IGA and IGSCC degradation modes has been developed.

Introduction

Nuclear steam generators are complex thermal, mechanical, and chemical systems. It is recognized that interdependences among thermal, mechanical, chemical, and metallurgical variables exist and exert strong and unique influences on the degradation of a given steam generator. It has been very difficult to account for these simultaneous interdependences analytically and make quantitative predictions of degradation due to small changes in operating temperatures, feedwater chemical additives, grain structure, or alloying element composition.

During the past four years, Northeast Utilities, Florida Power and Light, Arkansas Power and Light, and, more recently, Commonwealth Edison and Duke Power have worked together with C-E to develop future SAMs™ for nuclear steam generators. These models, which are supplied to the utilities, provide future predictions for the degradation characteristics of the respective steam generators as a function of planned operating and maintenance options.

The use of these models involves assembly of all relevant NDE, chemical, thermal, mechanical, and metallurgical data. The models also require input data on the future operating strategy. The results of the macroscopic component of the model include the average and the lower and upper bounds for the numbers of tubes in each region of the steam generator which will require plugging or sleeving due to a given type of degradation over the next few cycles. The microscopic component indicates if a specific type of degradation is sustainable in the steam generator. Thus, a utility can evaluate the impact of various operating strategies, such as utilizing new additives (e.g., morpholine), on degradation over the next few cycles without using the steam generator as an "experimental facility." Furthermore, model results can be employed in value-impact analyses to aid the utility in selecting among alternate strategies.

Before describing how the SAM™ technology (Ref.1) is applied to nuclear steam generators, a general summary of the analysis components is presented.

Analysis Philosophy

The goal of the SAM™ analysis is to predict the future performance, degradation, and availability of a nuclear steam generator. It is readily recognized that these three characteristics are strongly interrelated. To reflect these intrinsic interrelationships in a predictive analysis, the SAM™ technology concept was developed (see Figure 1). The performance component of the analysis sequence describes local thermal, hydraulic, and chemical conditions which exist throughout the steam generator. These conditions, based on anticipated future operational plans and constraints, are used as input to this part of the "microscopic" analysis sequence. The degradation component of the sequence combines the local conditions supplied by the performance component with analytic expressions relating the relevant degradation indices to the local conditions which drive the degradation.

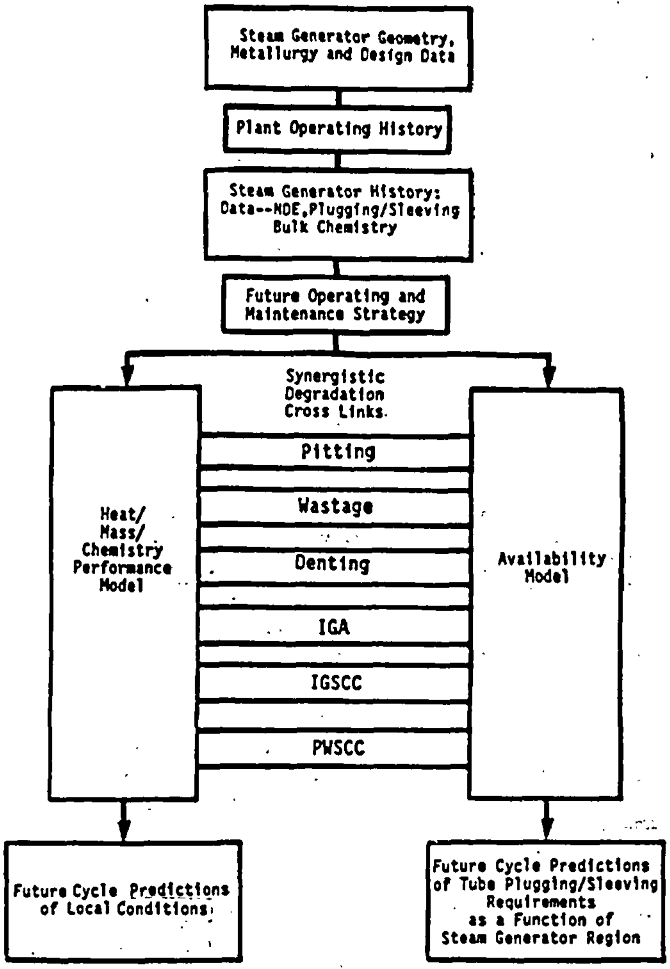


FIGURE 1. Nuclear Steam Generator Future Strategy Assessment Model (SAM™)

In the steam generator model, the microscopic analysis indicates if the degradation process is possible at a given location. If this is the case, the macroscopic analysis is utilized to determine propagation to undefected tubes and penetration of previously as well as newly defected tubes. These indices can be formulated as general corrosion material depletion per unit time as well as depth of localized attack. Finally, for a given depletion rate one can calculate the time to reach a specific threshold, such as a plugging or sleeving limit. This latter analysis in the sequence constitutes the availability component. The major advantage of this methodology is that it provides a logical and self-consistent framework for a computer model based on inspection data, availability reports, experimental test results, and physical principles. The model can be used by management decision-makers, as well as operating and maintenance personnel, to enhance their predictive capabilities in a repeatable and analytically defensible fashion. As a result, they more effectively can deal with the problems outlined earlier.

Degradation of steam generator tubes results from several corrosion phenomena, principally pitting, wastage, denting, IGA, and IGSCC. Most occur on the relatively cooler secondary sides of the tubes in or near cracks, crevices, and support structures, where deposits accumulate over time and provide sites for the concentration of corrosive impurities present in the secondary water.

Application of the Methodology to Pitting and Wastage - Millstone Unit II

The application of the SAM™ methodology to chemically initiated steam generator tube degradation has been focused on pitting, wastage, IGA, IGSCC, and denting. The performance component of the analysis supplies the local thermal-hydraulic and chemical conditions which exist throughout the steam generator. This function has been provided in part by the ATHOS™ computer code developed under EPRI. The chemical part of the performance model has utilized ratios of various bulk coolant impurities and computer codes such as the STAT-4™ chemical concentration code. Local thermal-hydraulic and chemical conditions can be determined for various regions within the steam generator (see Fig. 2).

The next analytic component involves the various degradation processes being addressed (pitting and wastage initially were modeled for Millstone II). The phenomena were found to be highly localized in the sludge deposit on the tube sheet. Statistical analyses of the relevant NDE data showed correlation between the height to which a tube was covered by sludge and the presence of defects. The next step in developing the degradation model for pitting and wastage was to determine the condition on the microscopic scale in each region necessary for pitting and wastage to occur. Tube-pull data indicated that specific chemical elements and compounds existed at pitting and wastage sites.

Combining this information with Pourbaix diagrams, illustrated in Figure 3 (such diagrams relate pH to corrosion potential for simple reaction products), the pH ranges required to initiate and sustain these two forms of degradation were identified. The ATHOS™ and STAT-4™ computer codes were employed to determine the pH that could be achieved locally at the tube walls. The critical pH was determined to be achievable only in tube sheet regions where at least a specified minimum sludge thickness could accumulate.

The final analytic element, the local concentration model, determines if the theoretically necessary conditions at the tube surface can actually occur. For the pitting and wastage phenomena, the model was prepared to examine the concentration mechanisms within the sludge surrounding a typical tube. Calculations of permeability, boiling point elevation, and mass transfer rates led to a transient model at this interface for the concentration of H_2SO_4. The results of this effort verified that the

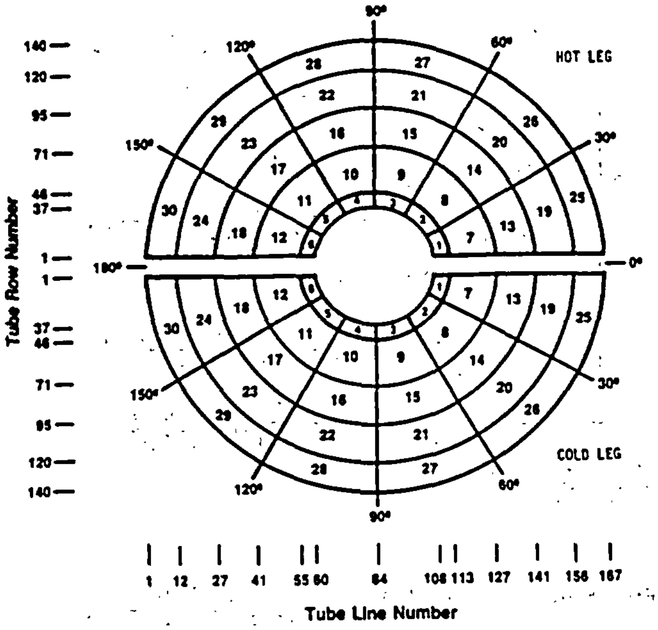


FIGURE 2. Regionalization Scheme for Steam Generator in Vicinity of Tube Sheet

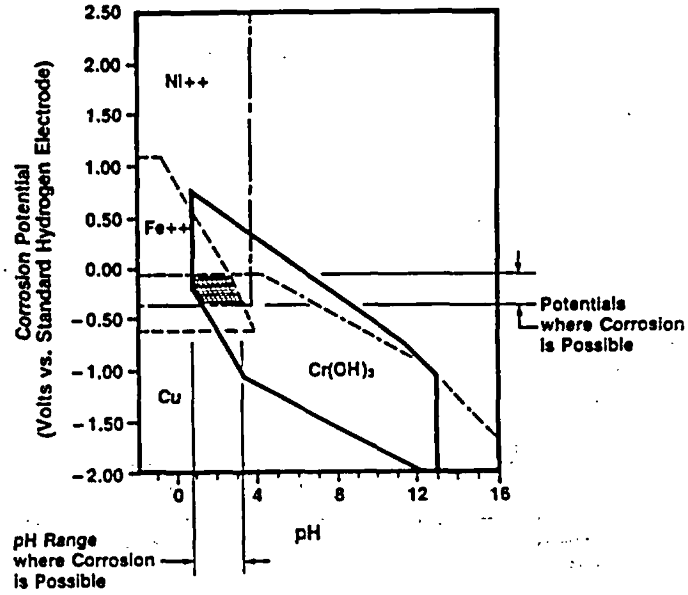


FIGURE 3. pH-Corrosion Potential Range where Pitting is Possible (Shaded Region)

theoretical concentrations necessary to initiate and maintain pitting on the cold leg and wastage on the hot leg were indeed possible. Furthermore, the thickness of sludge required to achieve these concentrations was only on the order of 10 mils, thus lending support to the concept of a sludge "collar" being sufficient to cause these forms of degradation even if a sludge "pile" did not exist or had been removed mechanically as via sludge lancing. The time-dependent modeling capability of the local concentration model was found useful in evaluating the length of time a given chemical imbalance could exist (e.g., due to a condenser leak) before the concentration of corrodent at the tube surface reached or exceeded the threshold for degradation (see Figure 4).

On the macroscopic scale, two different statistical models were developed to assess the progress of these modes of degradation. Borrowing from the terminology of public health, the rate of the spread of degradation to previously undefected tubing was termed "morbidity." Similarly, the rate at which previously or newly defected tubes degraded to the point where plugging or sleeving became necessary was termed "mortality" (Ref.2).

This statistical analysis is called the "macroscopic" part of the analysis because it utilizes the NDE data and extrapolates over several fuel cycles of operation. Figure 5 illustrates typical results for a specific steam generator region. The morbidity and mortality rate evaluation form the availability component of the SAM™. Pitting and wastage, addressed in the SAM™ for Millstone II steam generator predictions with and without chemical cleaning, were compared with actual inspection data in Figure 6. In this figure, for clarity, region specific results on the cold side were combined, as were the hot side results.

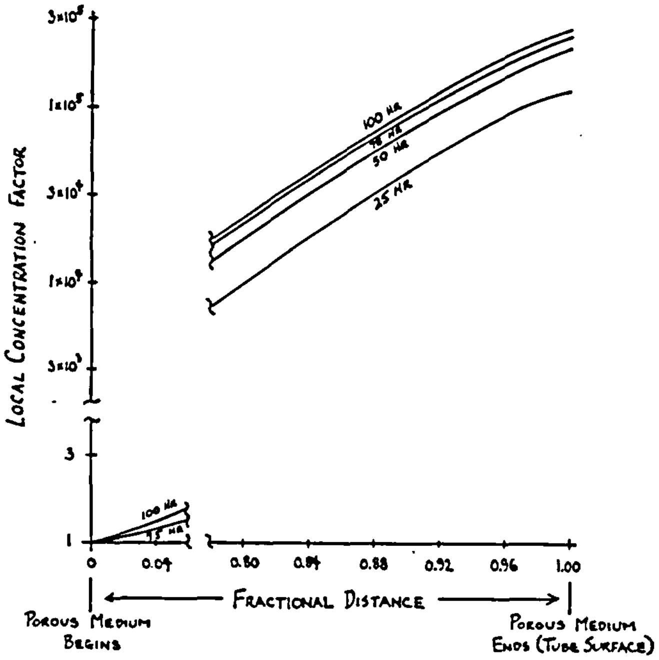
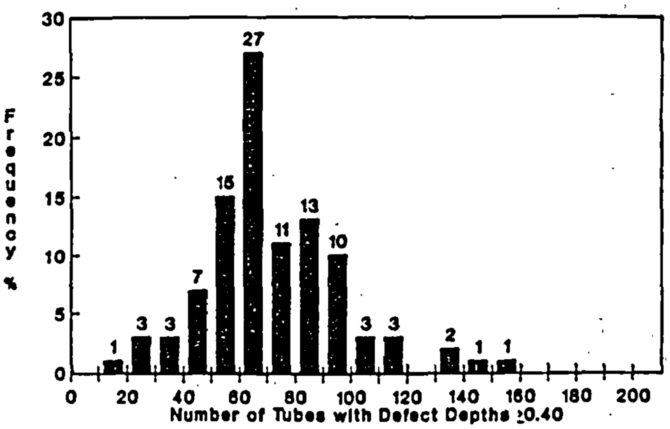


FIGURE 4. Local Concentration Model Results for One Millstone II Cold Leg Steam Generator Region



(NOTE: Region 15 Contains 324 Tubes)

FIGURE 5. Macroscopic Model Output for a Typical Region after Two Future Cycles of Operation

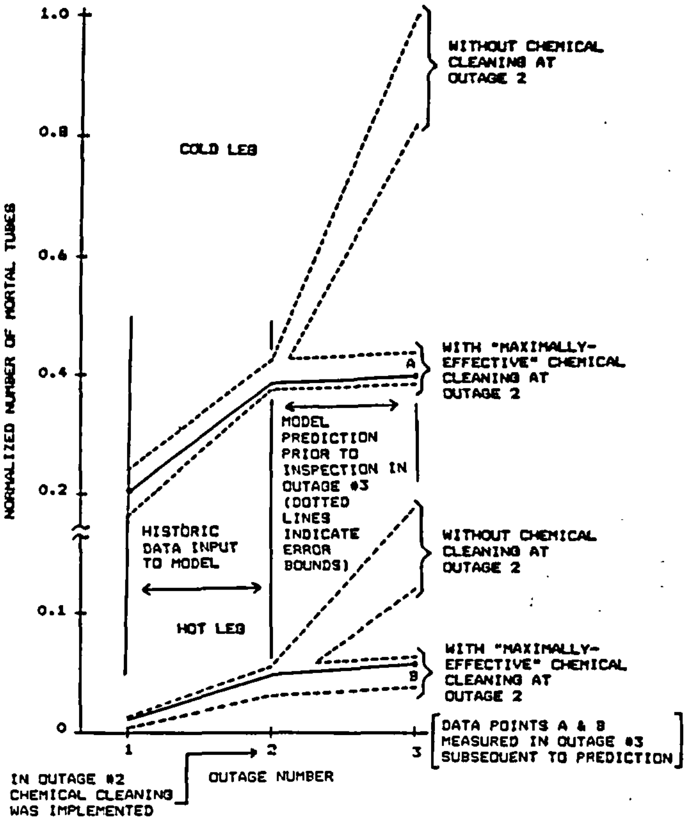


FIGURE 6. Observed (Solid Line) vs. Simulated (Dashed Line) Numbers of Mortal Tubes (Range Limits Only for Simulated Values) for Millstone-II per SG Leg for Three Consecutive Cycles

TABLE 1

Steam Generator Metallurgy, Service Conditions, and Susceptibility to IGA/IGSCC: Westinghouse vs. Combustion Engineering

	Westinghouse	C-E
Anneal Temp. (of Tubing)	1650 - 1700°F* (~900 - 925°C)	1850 - 1900°F (~1010 - 1040°C)
Carbides	Intragranular	Intergranular
Strength	High	Low
Grain Size	Fine	Coarse
Presumed Corrosion Properties	Greater Sensitivity to Caustic Attack/ Pure Water Attack	Greater Sensitivity to Acid Sulfate Attack
Cooling Water of most affected plants	Fresh Water	Sea Water

* This specific range of annealing temperatures is descriptive of Westinghouse tubing susceptible to IGA/IGSCC although some plants use tubing annealed at higher temperatures

IGA/IGSCC - Zion Units I, II; St. Lucie Unit I

The IGA and IGSCC degradation phenomena have been modeled for both C-E and W steam generator designs, for which they exhibit differences. Table 1 delineates the more obvious differences in metallurgy and potential secondary water contaminants.

The difference in presumed corrosion susceptibilities for C-E and W designed steam generators toward IGA/IGSCC has led to the development of separate degradation models for each type of generator. The C-E and W IGA/IGSCC degradation models will be discussed as separate and distinct analysis components.

The W IGA/IGSCC microscopic degradation model focuses on the crevice liquids that can accumulate and continue to exist within the deep tube sheet crevice, such as those in the Zion steam generators. These crevice liquids are governed by a number of factors chemical and physical, including the available superheat, solubility limits, and caustic buffering agents. These considerations led to a focus on the solution sodium acetate which, although it is not a corrodent, maintains the necessary liquid phase within the crevice to allow corrosion to take place. Carbon dioxide and its precursors, bicarbonates and the decomposition products of organics, have been proposed as corrodents for W steam generators. Thermodynamically, carbon dioxide should attack elemental iron in Alloy 600 at the grain boundaries because iron tends to concentrate at the grain boundaries (see Figure 7). Although this is theoretically possible, a physically-based local concentration model is required to see if the theoretical concentrations can exist in the deep tube sheet crevice of a real steam generator with all the thermal, hydraulic, diffusion, and chemical kinetic limitations.

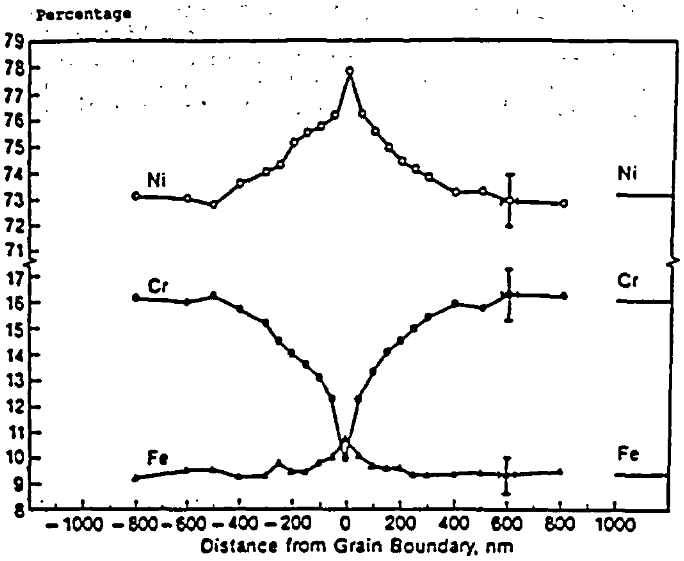


FIGURE 7. Elemental Distribution Across Alloy 600 Grain Boundaries, Mill Annealed (Ref. 3)

The C-E IGA/IGSCC model focuses on the crevice liquids that can accumulate and continue to exist within crevices between the steam generator tubes and the eggcrate and drilled support structures. These studies also indicate that the attack of acid sulfate is preferentially focused on the grain boundaries because of the normal depletion of chromium at these sites, as well as additional chromium depletion associated with intergranular carbide formation. Chromium appears to aid repassivation of tubing surfaces; since the rate of repassivation is a function of temperature, cooler surfaces may actually be more susceptible to IGA/IGSCC. Indeed, cold leg surfaces at St. Lucie I appear to have experienced greater defect growth than the hot leg, which presumably repassivates more rapidly. Thus, the theoretical/experimental basis for understanding the degradation exists. A local concentration model has been prepared to verify that the required thermal-hydraulic conditions, diffusion, and chemical kinetics exist within the steam generator environment. Again, this local concentration model is useful for determining the susceptible regions of the steam generator. The macroscopic model has been utilized to determine the morbidity and mortality for tubes in the susceptible regions.

Conclusion

Application of the complete SAM™ methodology utilizes many sources of information specific to the steam generator for which future performance, degradation, and availability are to be predicted. The detailed modeling of the degradation processes incorporates the complex interrelationships driving the phenomena. Thus, the sources of information can be utilized in a logically consistent manner. For a nuclear steam generator, the SAM™ methodology can address questions related to the feasibility of implementing operation and maintenance decisions, such as those related to bulk chemistry, thermal-hydraulics, chemical additives, and outage planning.

Acknowledgement

The modeling efforts discussed herein, as well as many ancillary analyses and research studies, could not have been made possible with the fine support provided by Dr. Raymond H.V. Gallucci, David B. Scott, Dr. Thomas A. Beineke, and Robert M. Rentler of C-E.

References

1. F. Berte', J. Klisiewicz, and E. Schwarz, "Computer Models, Methodology Help Predict Plant Behavior," 1987 Electric Utility Planbook, McGraw-Hill, New York, pp 11-15, 1987.

2. R. Gallucci, B. Woodman, and J. Klisiewicz, "Statistical Prediction of Tube Pitting and Wastage in Nuclear Power Plant Steam Generators," Proc. 1988 Joint ASME/ANS Nuclear Power Conference (Session on Advances in Steam Generator Operations and Maintenance), Myrtle Beach, South Carolina, April 17-20, 1988.

3. E. L. Hall and C. L. Bryant, "The Microstructure Response of Mill-Annealed and Solution Annealed INCONEL 600 to Heat Treatment," Metallurgical Transactions A, 16A, 1225. (1985).

LIFE PREDICTION FOR BWR INTERNALS

J. P. Higgins
GE Nuclear Energy

Abstract

Evaluation of life extension potential for Boiling Water Reactors (BWR) in the U.S. shows that plant lifetimes significantly greater than 40 years are attainable. Longevity of the reactor pressure vessel (RPV) and its internal components is of prime importance if operation beyond the 40-year design life is to be attained. Although there is a body of data available from RPV surveillance and ISI programs for life assessment of the reactor vessel, meaningful ISI data for RPV internal components are meager. Reasonable estimates of residual life must be an integral part of any successful life extension plan in targeting components for aging prevention, special surveillance, and refurbishment or preplanned replacement.

Operating experience and laboratory testing has shown that for BWR vessel internals, the principal life limiting mechanisms are intergranular stress corrosion cracking (IGSCC), irradiation assisted stress corrosion cracking (IASCC), and fatigue. Significant plant-to-plant variation in component design features, duty cycles, and material characteristics plus plant unique operating characteristics such as water chemistry, component fluence levels, and in-service inspection (ISI) results must also be considered when making a assessment of residual life.

This paper identifies several life prediction techniques which have been applied to BWR vessel internals and presents applications of these techniques.

Life Limiting Mechanisms

Historical data has shown that for BWR austenitic stainless steel (S/S) internal components the principal degradation mechanisms are IGSCC, IASCC, and fatigue.

Intergranular Stress Corrosion Cracking (IGSCC) of austenitic stainless steels in high temperature/high purity BWR water is a well documented phenomenon in which the occurrence of cracking is related to the concurrent interaction of high tensile stress, an oxidizing environment, and a susceptible material. Historically, a susceptible material is characterized by a sensitized microstructure related to precipitation of chromium carbides in grain boundaries with an attendant chromium depletion in the adjacent matrix. IGSCC in BWRs typically occurs in stainless steel recirculation system piping and reactor vessel safe-ends. More recently, limited incidents of IGSCC in welded and creviced RPV internal components have been observed.

Irradiation-assisted stress corrosion cracking (IASCC) is a form of IGSCC that has infrequently been observed in reactor internal components of non-sensitized S/S. IASCC was first observed in S/S fuel rod cladding and later in control blade absorber tube cladding where relatively high stresses are achieved due to fuel or B_4C swelling within the tube. More recently, cracking has been observed in structures subject to nominally lower stresses, such as the control blade sheath and handle, neutron source holders, and both source range monitors (SRMs) and intermediate range monitors (IRMs). Based on available field and laboratory data, a "threshold" fast neutron fluence ($E > 1$ MeV) of about 5×10^{20} n/cm^2 appears to exist for IASCC of annealed austenitic stainless steel and Alloy 600 in the more highly stressed components and 2×10^{21} n/cm^2 for lower stressed components.

Fatigue cracking occurrences of BWR internal components, although well understood, have always occurred as the result of loadings not anticipated. The most notable example is that of high-cycle thermal fatigue cracking of feedwater spargers/thermal sleeve assemblies due to bypass leakage between the feedwater nozzle and the thermal sleeve. Other fatigue failures in jet pump sensing lines and steam dryer assemblies have also been observed. In each case failure prediction without prior knowledge of the cyclic loading condition which produced the failure was not possible.

Bases of Life Estimates

The component life estimates presented here are for reference cases only and are not broadly applicable to all BWRs. Plant-to-plant variations, such as component design, fabrication details, and plant unique differences in important variables such as water quality and operating history require plant-specific evaluation for component life estimates.

The component life estimates given in this report are based on the following factors:

- Extrapolation From Field Experience

  IGSCC has occurred in sensitized or creviced stainless steel safe ends in many operating plants within five to ten years from startup. Similarly, IGSCC has occurred in creviced Alloy 600 shroud head bolts and in-core neutron flux monitors.

- Delphi Approach

  Where there is no established data base on field failures and the most likely failure mechanism is known, analytical predictions based on the Delphi technique have been used to form life estimates. In some cases, laboratory data has been used to form a basis for judgment for life estimates.

- On-Line Monitoring

  A variety of on-line monitoring techniques have been developed, validated, and applied in operating plants to predict the performance of reactor components.

Life Prediction Methods

Extrapolation From Field Experience

Field performance history of certain creviced in-reactor internals components has revealed a clear correlation between BWR coolant conductivity and the propensity for IGSCC. BWR components that have demonstrated a strong correlation between coolant conductivity and the initiation of IGSCC included creviced Alloy 600 shroud head bolts, creviced and irradiated stainless steel intermediate and source range monitor (IRM/SRM) dry tubes.[1] IGSCC initiation in these components was significantly higher in plants that had been operating with relatively high lifetime coolant conductivities.

Alloy 600 Shroud Head Bolts. Creviced Alloy 600 shroud head bolt cracking was first discovered in January 1986. The UT examinations conducted at several BWR plants revealed shroud head bolt cracking in a number of other plants. Cracking occurred in the creviced region in the vicinity of a weld joining the Type-304 SS collar to the Alloy 600 bolt. Metallographic evaluation showed that the microstructure was characteristic of mill-annealed Alloy 600 and confirmed that the cracking mechanism was IGSCC.

Shroud head bolt inspection results are presented in Figure 1 with the percentage of bolts containing UT crack indications (normalized to time on-line) plotted as a function of lifetime average water conductivity. Each data point represents inspection results for one BWR plant. The incidence of shroud head bolt cracking creases linearly with increasing lifetime average water conductivity above a threshold value of approximately 0.25 μS/cm.

IRM/SRM Dry Tubes. Stress corrosion cracking in annealed and irradiated IRM/SRM dry tubes was first discovered through visual inspections in 1984. Cracking has subsequently been observed in dry tubes at other BWR plants. Dry tube cracking is generally attributed to crevice accelerated IGSCC and/or IASCC. Although dry tubes were annealed before fabrication, and stresses are considered negligible, thick oxide formation in the crevices appears to have served as a significant source of stress (oxide wedging). Fluence levels in cracked dry tubes ranged from $5x10^{21}$ to $1x10^{22}$ n/cm^2 (E>1 MeV).

Visual inspection data for dry tubes were evaluated in the same manner as shroud head bolts. Figure 2 shows the percentage of cracked IRM/SRM dry tubes (normalized to time on-line) plotted as a function of plant lifetime average water conductivity. A correlation between the propensity for SCC initiation and BWR water conductivity, similar to that observed in the shroud head bolts (~ 0.25 μS/cm), was also apparent for the IRM/SRM dry tubes.

Delphi Approach

The Delphi approach to component life prediction relies on performing a review of key plant operating history parameters (coolant chemistry, plant cycles, chemistry transients) and a detailed assessment of component specific features (i.e., design, materials, fabrication, installation, service, and ISI data). This information, supplemented with operating service histories of the same or similar components, is assembled for review by an expert panel. Life limiting mechanisms are identified with focus on key component areas (i.e., welds, crevices), and an estimate of probability for time to crack initiation and time

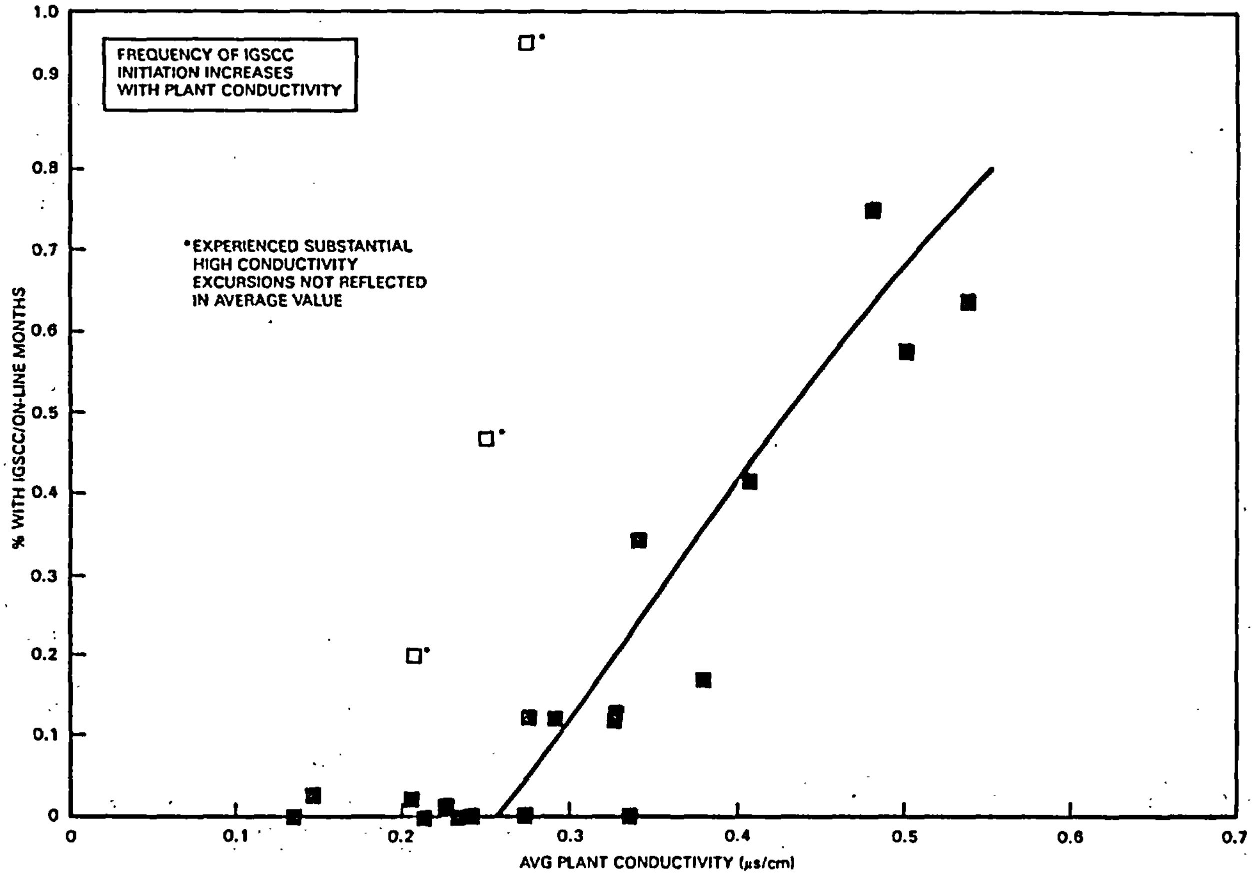


Figure 1. Correlation for Shroud Head Bolt Cracking and Plant Conductivity

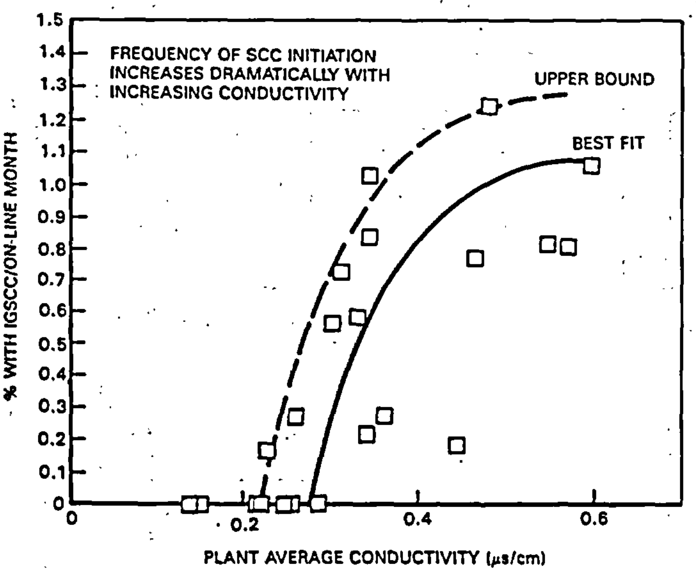


Figure 2. Correlation for IRM/SRM Dry Tube Cracking and Plant Conductivity

to repair or replace is generated by consensus of the expert panel.

These life estimates may also be updated with inspection data, should it later become available, as was the case for the CRD housing. This incorporation of field experience is accomplished with Bayes' theorem.

CRD Housings. There are no known occurrences of IGSCC in CRD housings although high quality ISI data is meager. In view of the known high degree of susceptibility of Type 304 stainless steel weld HAZs in other BWR components, the non-cracking in CRD housing weld HAZs is attributed to the low stress levels at this location. Based on these data and the consensus of an expert panel, the predicted life of a single housing can be said to be longer than 80 years. This life estimate was updated with ISI data from one BWR where 97 housing weld heat affected zones (HAZs) were ultrasonically examined after approximately 15 years of operation and found to be crack free. However, this prediction must be adjusted to account for the fact that the BWR typically contains 137 to 185 CRD housings.

Figure 3 shows estimated life for any given housing based on the updated probability which includes the more recent favorable CRD housing ISI data.[2] Figure 3 also shows the estimated time to failure for the first housing of the plant total population.

On-Line Monitoring

The best predictor of reactor component life is operating experience. For example, while industry codes and standards provide guidance to account for well understood phenomena such as fatigue, only after the fact field data has shown where fatigue has become a life limiting mechanism. Ideally, what is required is an on-line monitoring methodology based on a combination of crack monitors, environmental sensors, and a mechanistically based model capable of quantitative life prediction.

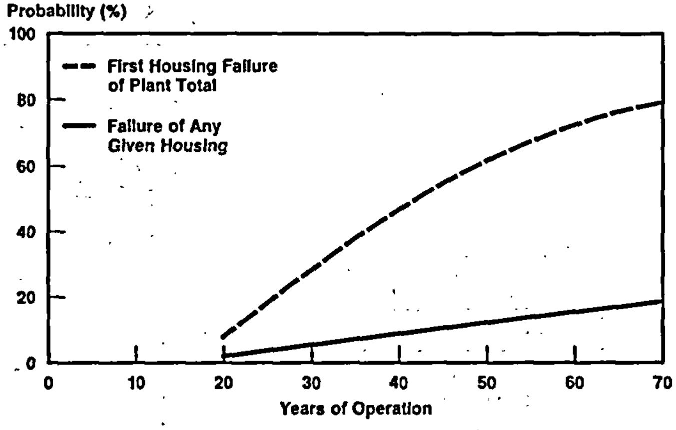


Figure 3. Predicted Failure Probability for CRD Housing

Developments resulting from work performed at General Electric and supported partly by the Electric Power Research Institute allow on-line monitoring of crack growth, water quality and stress cycles to confirm design margins and assure structural integrity throughout the extended life of the plant.[3] Examples of such monitors include application of the reversing DC potential technique to measure crack growth, ion chromatography and electrochemical potential sensors to detect coolant impurities and control conductivity.

Conceptually, the crack growth monitor permits the state of damage at specific plant locations to be established either by direct monitoring of plant components or indirectly by extrapolation from a "reference" component such as a precracked compact tension fracture mechanics specimen.

Figure 4 presents a typical materials monitoring system developed by General Electric for BWR application. This system incorporates a crack advance verification system (CAVS) and a pipe crack monitor (PCM). In the CAVS, precracked specimens are placed within an autoclave and water is routed to the specimens from recirculation piping or other primary system pipe. The specimen is mechanically loaded to develop a specified stress intensity at the crack tip and crack advance is remotely monitored to a precision of 25 microns. CAVS data, combined with an interpretive model relating crack growth rate to stress intensity, provides the means for realistic prediction of remaining component lifetime.

Figure 5 gives typical results from CAVS for a Type 304 S/S RPV safe end. The CAVS growth prediction is well below the bounding crack growth evaluation based on plant lifetime average water chemistry. The final crack depth at the end of a fuel cycle is below the allowable depth based on providing an ASME Code margin of 3 on stress and an additional factor of 1.5 on crack depth uncertainty. These results confirmed that sufficient structural margins were maintained at an operating BWR so that a special midcycle examination was not required.[4]

References

1. K. S. Brown and G. M. Gordon, "Effects of BWR Coolant Chemistry on the Propensity for IGSCC Initiation and Growth in Creviced Reactor Internals Components," Third International Symposium

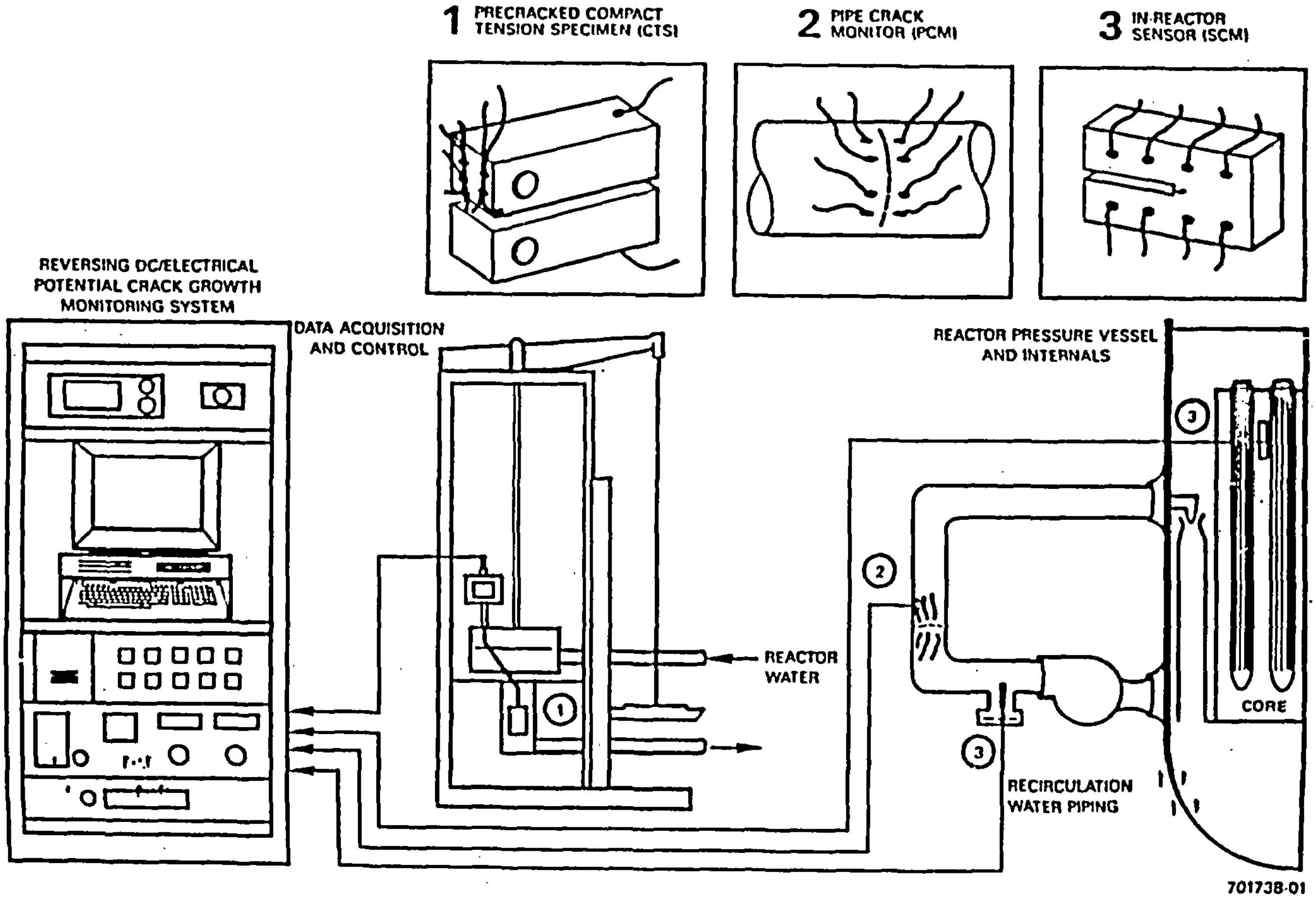


Figure 4. Real Time BWR Materials Monitoring

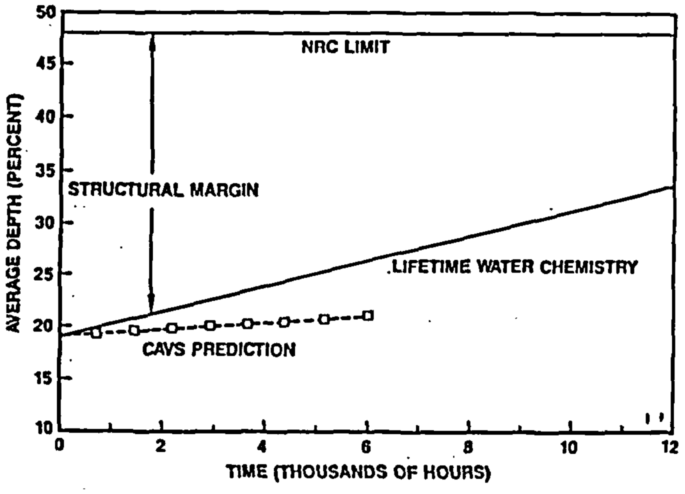


Figure 5. Predicted Crack Growth in Safe End

on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors, Grand Traverse Resort Village Traverse City, Michigan, August 30 - September 3, 1987.

2. J. P. Higgins, "Longevity of BWR Reactor Internal Components," Proceedings of Seminar on Nuclear Power Plant Life Extension," Alexandria, Virginia, August 25-27, 1986.

3. F. P. Ford et al., "On-Line BWR Materials Monitoring and Plant Component Lifetime Prediction," ANS Topical Meeting on Nuclear Power Plant Life Extension," Snowbird, Utah, August 1988.

4. R. J. Brandon and P. P. Stancavage, "For BWR Longevity, Plants Are the Best Teachers," Proceedings of An International Symposium On Safety Aspects of the Aging and Maintenance of Nuclear Power Plants Organized by the International Atomic Energy Agency and Held in Vienna, June 29 - July 3, 1987.

FATIGUE USAGE FOR LIFE EXTENSION OF BOILING WATER REACTOR VESSELS

K. Mokhtarian

INTRODUCTION

Most of today's Boiling Water Reactor Vessels were designed and analyzed in the 1960's and early 1970's. (Chicago Bridge & Iron Co. was a major supplier of BWR vessels during that period and this paper reflects the design experiences of that company.)

There has recently been a great deal of interest in extending the life of nuclear plants beyond their original design life. The originally calculated fatigue usage factors, for a specified design life of 40 years, were quite high in some areas. As utilities try to extend the life of their nuclear plants, questions arise regarding the validity of the original analyses and the remaining fatigue life of vessel components. The original analyses were based on a number of conservative assumptions. With today's better understanding of the vessel operating conditions and with better analysis tools available, a much more accurate estimate of a vessel's fatigue life can be calculated. This paper will point out some areas of conservatism in the original calculations and some means of improving on the accuracy of calculations.

CALCULATION OF FATIGUE USAGE FACTORS

The original fatigue usage factors were based on the number of cycles specified for various normal, upset, emergency and faulted conditions. The specified number of cycles is a conservative estimate of what is expected during a 40 year design life. For analysis purposes, the reactor pressure vessel was broken down into a number of components, such as nozzles, support skirt, closure flange, and shroud support. For each component, and for each specificed condition, the temperature distribution as a function of time had to be calculated. The stresses due to pressure, temperature, and other loadings were then calculated, as a function of time. Based on the range of stresses, a partial fatigue usage factor was calculated for each type of cycle. If the sum of all such partial usage factors did not exceed 1.0, the design of the component was acceptable. Section III of the ASME Code was used for design acceptance criteria.

Since accurate calculation of temperature distributions, stress ranges, and the resulting fatigue usage factors was quite time consuming, conservative assumptions were made, as long as such assumptions did not result in the calculated fatigue usage factors exceeding 1.0. With such conservative assumptions, the calculated usage factors for some components were rather high. The following is a summary of some of the relatively high usage factors that were calculated for a typical reactor vessel, by Chicago Bridge & Iron Company, in the early 1970's.

Component	Fatigue Usage Factor
closure studs	0.89
head flange	0.92
feedwater nozzle safe end	0.92
feedwater nozzle thermal sleeve	0.87
core spray nozzle safe end	0.49
recirculation outlet nozzle at junction to shell	0.48
steam outlet nozzle at junction to shell	0.78
core differential pressure and liquid control nozzle	0.61
refueling bellows	0.47
stabilizer bracket at junction to shell	0.72

AREAS OF IMPROVEMENT

There are a number of areas in which improvements can be made to the original calculations. Based on available operating history, the portion of fatigue life consumed in the past can be calculated and the remaining fatigue life can be estimated.

The ASME Code Rules do not presently provide for in-service fatigue life evaluations. However, a number of organizations including ASME are looking at developing rules for life extension of nuclear vessels. Any life extension rules will have to address the fatigue usage of components.

Some of the areas in which the accuracy of the original analyses can be improved are listed below:

Operating Cycles

The operating conditions and the number of cycles of each condition can be estimated much more accurately now. The log of operating history will provide an accurate basis for calculating the portion of fatigue life already consumed. This operational history will also provide for a much more dependable estimate of future operating cycles.

Calculation of Surface Film Coefficients

The temperature distributions and in turn, thermal stresses are a function of surface heat transfer film coefficients. With better understanding of operating conditions and more accurate methods available for calculating film coefficients, the accuracy of results can be improved.

Heat Transfer Calculations

Almost all heat transfer calculations were based on 2-dimensional (axisymmetric) idealized models. In reality, most components are 3-dimensional in geometry. Computer programs are now available to readily perform 3-dimensional heat transfer analyses. Moreover, these programs are coupled with finite element stress analysis routines, so that the calculated temperatures can be directly utilized for stress analysis. When the original analyses were performed, the temperature distributions had to be output at a predetermined number of times; the resulting temperature distributions were evaluated for determining "worst cases," and then these worst distributions were linearized at a number of sections for input into shells of revolution analysis program. It took a great deal of judgment to pick the "worst cases," and the linearization and idealization of temperature distributions could have resulted in less accurate results than could be accomplished today. With today's coupled finite element analysis and extensive pre- and post-processing capabilities, a more accurate estimate of thermal stresses is possible.

Stress Calculations

Most of the stress analysis of components was performed by a shells of revolution computer program. Again, the details had to be idealized for axisymmetric modelling, and in some areas the thicknesses were such that using thin shell theory could have resulted in inaccuracies. With today's finite element analysis tools, much more dependable stress analysis can be performed. The use of three dimensional finite element analysis will provide a much more accurate estimate of local discontinuity stresses and peak stresses. Pre-and post-processing capabilities are available to allow for calculation of stresses of different categories and for properly categorizing stresses for ASME Code evaluation. With today's tools, the stresses due to various sources can be summed in an exact manner and then stress ranges calculated. For the original analysis, the stresses due to some loads (such as nozzle reactions) were calculated by hand and added in a conservative manner (worst of times and locations combined) to pressure and thermal stresses calculated by the computer program.

Fatigue Calculations

In order to perform fatigue analysis, the calculated stresses were multiplied by a fatigue strength reduction factor (based on stress concentration factor) in order to arrive at "peak" stresses. These factors were conservatively calculated by use of various references which contained stress concentration factors for regular geometries. The actual discontinuities had to be idealized to fit one of the published details. Today, better estimates of stress concentration factors are available. Or, such factors can be determined for the actual detail by use of finite element computer programs.

The ASME Code has contained curves for determining the allowable number of cycles for a certain range of peak stress. Although these Code curves have been little revised since the 1960's, much better material data is now available for determining the allowable number of cycles. A great deal more is known about the effects of operating conditions on fatigue properties of materials. The ASME Code fatigue curves contain safety factors of 2 and 20 on stress range and number of cycles, respectively, based on test results of new polished specimens tested in air. These factors are to account for such effects as deterioration of materials with service, the residual mean stresses, scatter in fatigue data, and size effects. The ASME Code Committees are working on replacing or revising the fatigue curves, based on the extensive amount of data which has been published within the last decade. Some of the material damage mechanisms that have been investigated are the effects of corrosion and erosion, the effects of irradiation, thermal aging, hydrogen damage, and others. In general, means are available for much more reliable fatigue analysis than the present Code curves.

Since the object in the original analysis was to show that the cumulative fatigue usage factor at each point was less than 1.0, cycles of different stress range were sometimes lumped together in a conservative manner. Any accurate determination of fatigue usage will have to treat each type of cycle individually. Post-processing capabilities are available to readily accomplish this.

CONCLUSIONS

Much better material data and analysis tools are available today than were used for original design of BWR vessels. In general, the more refined analysis would provide a more accurate assessment, and obviate the need for making conservative assumptions.

For those plants for which life extension is contemplated, a recalculation of fatigue usage factor at a number of critical points will be needed. The most critical points, from a fatigue standpoint, can be identified from the original stress reports.

While this paper discusses BWR vessels only, a similar situation exists with extrapolating the operating life of other reactor types. A refined fatigue analysis will be needed to estimate the fatigue life consumed and the fatigue life remaining for the reactor vessel.

IMPLICATIONS OF THERMAL AGEING OF CAST AUSTENITIC STAINLESS STEEL AND WELDMENTS

T. Hardin, W. Pavinich, and W. Server

Abstract

Cast austenitic stainless steel (CSS) pipes, elbows, valves, and pump casings used in the LWR primary coolant pressure boundary are subject to gradual embrittlement - loss of fracture toughness - when exposed to service temperatures (280-320°C, 536-608°F). Weldments are also subject to embrittlement. The present inability to accurately quantify the degree of toughness degradation over the lifetime of the LWR, coupled with indications from some studies that the degradation is dramatic, has raised concern for the integrity of aged CSS components and weldments. The concern is that as embrittlement progresses and fracture toughness decreases during long-term service exposure, defects or flaws in the material will grow to a critical size and the component will fail. Therefore, there exists a need for further research and plant-specific tracking of this material degradation. The authors have conducted a literature search and expert review of the ageing phenomenon, including potential ways to monitor its progress, and have produced a three phase program for monitoring the thermal embrittlement at nuclear plants. The thrust of the program is to screen CSS components and weldments on the basis of their susceptibility to ageing (a function of material composition and service temperature) and identify a small but fully representative set for which an active monitoring program is economical and feasible. The screening program will identify components that are most susceptible: components with high ferrite, carbon, chromium, and molybdenum contents. The program is phased in order to firmly establish its technical basis for eventual regulatory acceptance.

Introduction

This paper is a result of work to develop a program plan for monitoring thermal embrittlement of primary coolant cast stainless steel components in Tennessee Valley Authority's nuclear power plants. The work included a review of available literature and industry experts to enhance the understanding of the thermal embrittlement phenomena. Additionally, candidate test methods to monitor thermal embrittlement of cast stainless steels were considered for evaluation.

Cast austenitic stainless steel (CSS) is a duplex structure alloy consisting of two metal phases, austenite and ferrite. The alloy is predominately austenitic, but the ferrite (10% to 25% by volume) significantly improves strength, resistance to corrosion, and weldability. These properties make CSS an excellent material for use in a nuclear primary containment boundary, and CSS is used to varying extent in all nuclear power plants, usually in the form of statically cast reactor coolant pump casings or statically or centrifugally cast primary loop piping. The only grades used for nuclear applications are CF8, CF8M, and CF8A (1). Therefore, unless otherwise noted, the discussion of thermal aging in this report refers to the phenomenon as seen in CF8/8A/8M.

CF8 and CF8M are chemically equivalent to (wrought) Types 304 and 316, respectively. CF8A is a strengthened CF8 (CF8A has slightly more Si, giving it a yield strength about 5 ksi higher) and is the most common CSS grade in nuclear piping systems.

In the course of conducting the literature search and expert review, the authors looked at the issue of welds in CSS materials and their susceptibilty to thermal ageing. The available data indicate that weldments are also subject to thermal ageing and that the effect could be significant regardless of the parent material (wrought or CSS). Therefore, this paper includes a discussion of thermal ageing in weldments.

Background

Thermal Ageing Phenomenon

Thermal ageing of cast austenitic stainless steel has been recognized since 1965 but was thought to be significant only at temperatures (>425°C, 797°F) well above the LWR operating environment. Elevated temperatures activate the precipitation of various phases in the ferrite, causing the material to embrittle -- gain strength and lose fracture toughness. Recent evidence that activation of the embrittlement process starts at temperatures as low as 200°C (392°F) has resulted in a concern for the structural integrity of nuclear plant components fabricated of CSS and has stimulated some major research which is in progress but incomplete. The concern is that as embrittlement progresses and fracture toughness decreases during long-term service exposure, defects or flaws in the material will grow to a critical size and the component will fail. The purpose of this discussion is to summarize current understanding of the increase in yield and ultimate tensile strengths and a decrease in Charpy impact energy, fracture toughness (J_{IC}), and tearing modulus. In thermally aged CSS materials, fracture on the microscopic scale occurs in the ferrite by cleavage (even at fairly high temperatures) or phase boundary separation, although ductile tearing (microvoid coalescence) predominates in the austenite. Until the mid-1980's thermal ageing was poorly understood because the precise mechanisms of embrittlement were unknown. Recent research employing transmission electron microscopy (TEM), atom probe field ion microscopy (APFIM), and small-angle neutron scattering (SANS) techniques reveal that aging in the 280-450°C (536-842°F) temperature range is the result of five complex metallurgical processes (16,23,24):

1. Spinodal decomposition of the ferrite (fcc) (the spontaneous decay of initially homogenous alloy by diffusion up a concentration gradient) forming chromium-rich alpha prime (bcc) phase;
2. Nucleation and growth of the alpha-prime phase (Cr-rich);
3. Precipitation of G-phase (an fcc nickel silicide);
4. Precipitation of gamma-2 austenite; and,
5. Precipitation of spherical $M_{23}C_6$ carbides at the austenite-ferrite boundaries.

Note that these processes occur only within the ferrite or at its grain boundaries; austenite does not experience thermal ageing.

The common mechanism of embrittlement in any grade CSS is the spinodal decomposition of the ferrite, but embrittlement in CF8/8A/8M is enhanced by G-phase precipitation within the ferrite and $M_{23}C_6$ precipitation at the phase boundaries. The high carbon (C) content in CF8 accelerates the $M_{23}C_6$ carbide precipitation, and in CF8M both C and Mo accelerate the G phase precipitation (18). For these reasons, CF8 grades are significantly more susceptible to low-temperature thermal ageing than low carbon grades such as CF3.

Embrittlement of CSS is a complex function of both the temperature at which it is aged and the specific chemical composition of the material. Some of the relationships are straight-forward: the greater the ferrite content of the CSS, the greater the embrittlement; the greater the carbon content of the ferrite, the faster the embrittlement. The quantitative magnitude of each effect is difficult to ascertain, however, because of the effects of other chemical additives such as nitrogen which is known to play a significant role (5). Consequently, the rate and full extent of embrittlement over the lifetime of affected components are not yet fully established. Impact on plant safety, if any, has not been quantified. Current data indicate that the embrittlement is significant and variable, which implies a need for plant-specific tracking of this material degradation.

Current Research

Before quantitatively discussing the degradation of fracture toughness in CSS, it is important to describe the research programs which study this issue. Most of the reliable and available information concerning thermal embrittlement has only recently been published and is a result of the U.S. NRC's program to investigate the significance of in-service embrittlement of CSS in the LWR environment (8). Started in 1982, the program includes Charpy impact, tensile, and J-R curve tests on various statically and centrifugally cast heats of CSS aged up to 50,000 hours at 290-450°C (554-842°F) and is being performed by Argonne National Laboratory (ANL). Some previous data exist but must be viewed critically because of the invalid assumption, discussed below, upon which most earlier studies are based.

In an effort to predict long term, low temperature thermal ageing effects within a reasonable time, most studies aged CSS samples at very high temperatures (>400°C, 752°F) under the assumption that such exposure produces an embrittled condition equivalent to that obtained by longer time exposures at actual (relatively low) LWR operating temperatures. This method employs an Arrhenius extrapolation:

$$t = 10^P \exp\left[\frac{Q}{R}\left(\frac{1}{T} - \frac{1}{673}\right)\right] \qquad (1)$$

where t is the ageing time to reach a given degree of embrittlement at different temperatures, P is the ageing parameter which represents the degree of ageing reached after 10^P hours, Q is the activation energy, R is the gas constant, and T is the absolute temperature (5). Under such a relationship, end-of-life (40 year) embrittlement for LWR CSS materials could be estimated by ageing samples at 400°C (752°F) for somewhere between 10,000 and 60,000 hours (depending on material grade and service temperature).

Recent tests at Argonne National Laboratory (ANL) show that this Arrhenius extrapolation is not valid for all types of CSS materials because of differing microstructural evolutions at low and high temperatures (16). Stated simply, thermal ageing at 450°C occurs via different mechanisms than at 300°C. Studies which base their conclusions on the Arrhenius extrapolation, then, give misleading predictions with respect to the rate and extent of embrittlement and should be viewed with some circumspection.

Specifically, ANL found that the precipitation and/or growth of phase boundary carbides dominates the embrittlement behavior of high-carbon steels (CF8 and 8M) aged at temperatures greater than 400°C (752°F), whereas spinodal decomposition is the primary microstructural evolution at LWR temperatures (9). CSS aged at high temperatures embrittles quickly and reaches a fracture toughness plateau by 10,000 hours. CSS aged at lower (LWR) temperatures embrittles more slowly but experiences a continued decline in fracture toughness past 10,000 hours (27). Whether or not the slowly aged material fracture toughness end point is lower will not be resolved until the 30,000 and 50,000 hour aged samples are tested. Nevertheless, it is clear that studies which aged CF8/8M at 450 or 400°C (842 or 752°F) should not precisely extrapolate the behavior to the LWR environment.

Changes in Fracture Toughness- Cast Material

Many of the published studies report the embrittlement of CSS in terms of Charpy impact energy changes, and most reports show dramatic drops in impact energy for the types of CSS grades in use in the nuclear industry (CF8M and CF8A). Room-temperature impact strength can be reduced by 80% after ageing for 8 years at temperatures as low as 300°C (572°F) (8). This implies a significant loss of fracture toughness at typical PWR/BWR temperatures. However, Charpy impact data do not sufficiently characterize the toughness behavior. J-R toughness properties are also required (18).

Using J-R toughness data from their own ANL study and data published by Westinghouse (7), Framatome (6), and EPRI (10), Chopra and Chung demonstrate in Reference 8 that the J_{Ic} and tearing modulus (T) of aged CSS decrease, that the decrease is seen at 290 C(550°F) (upper shelf) as well as at room temperature, and that the drop in J_{Ic} is similar to the dramatic drop in Charpy impact energy. Even so, there is some opinion that the room temperature lower bound J_{Ic} may be as high as 100 kJ/m^2 (600 in-lb/in^2) (21). If this true, then perhaps CSS toughness levels do not drop to an aged level less than that of weldments, which would tend to alleviate the integrity issue. Weldments have long been known to possess much lower toughness than base metal without being thought to pose a significant compromise in integrity or safety (22). In fact, the weld lower bound has been used to develop an austenitic stainless steel screening and analysis procedure prescribed in the ASME Boiler and Pressure Vessel Code, Section XI, IWB-3640 (9). The embrittlement of parent CSS is not the only possible integrity issue, however. Thermal ageing of weldments may be of greater significance to integrity and should not be ignored.

Changes in Fracture Toughness- Weldments

There are few published data concerning the effect of thermal aging on weldments, and essentially all available data are based on accelerated ageing laboratory results. It is known that the effect of thermal ageing in welds is much less significant as a percentage change in toughness than the effect in cast steels. Because the unaged weld fracture toughness is so significantly below that of unaged CSS, however, there is the possibility that the end-of-life condition, though not dramatically changed, is unacceptable. This possibility deserves more study.

A 1983 study by Slama et al (6) using Charpy impact data concluded that the effect of ageing is not important for welds. A more thorough 1987 English study by Hale et al (18) performed J_R testing on CF3 and manual metal arc (MMA) weld metals and found that aged weld metal experiences a considerable reduction in resistance to stable crack growth, although the reduction is not as great as for the parent CF3 material. Unaged weld metal J_{Ic} at 300°C (572°F) was estimated to be approximately 100 kJ/m^2 (571 in-lb/in^2); after ageing at 400°C for 10,000 hours, J_{Ic} was reduced to 75 kJ/m^2 (428 in-lb/in^2).

The significance of 75 kJ/m^2 (428 in-lb/in^2) is that it is well below the lower-bound J_{Ic} value of 114 kJ/m^2 (650 in-lb/in^2) assumed for flux welds in the ASME Code (IWB 3640) evaluation of flaws in austenitic piping (9). It raises the possibility that some aged welds in LWR plants do not in fact meet the assumed toughness characteristics of the Code.

Reactor coolant pump (RCP) casing integrity is another concern. Given the use of welds in both Type E and Type F RCP casing fabrication, the unavoidable flaws introduced in the welding process, and the extreme difficulty of ultrasonic, liquid dye penetrant, and radiographic testing of CSS RCP casings to detect those flaws, there is a possibility of flaw growth to critical size if fracture toughness reaches such a low value (25,26).

Candidate Testing Techniques

Discussion

Fracture toughness degradation due to thermal aging is caused by microstructural transformations in the ferrite of the CSS (and, to a lesser extent, precipitates at the grain boundaries). The rate and degree of embrittlement are complex functions of the chemistry of the particular material and are not easily generalized. Embrittlement is different for each CSS grade (CF8A, CF8M) and is different for relatively minor chemical variances within each grade (i.e., nitrogen content). Because of the complexity, each CSS component is affected differently and can see its own unique toughness property shift. An effective and economical program to monitor thermal ageing would screen each CSS component in a plant on the basis of material chemistry and service temperature, establish a thermal embrittlement susceptibility hierarchy, and monitor those components which present a bounding or limiting condition. Such a program minimizes the number of plant components subjected to physical tests and is presented later. This section presents the testing techniques available to monitor those components should fracture toughness testing be required.

Material property testing of in-service components implies access to critical welds, pipes, and RCP casings in the primary coolant system. The sampling method should not damage the CSS component and necessitate costly repairs. To this end, then, there are two relatively quick, un-obtrusive and minimally- destructive methods of testing the CSS materials (pipes, valves, and pump casings):

a. miniaturized/subsized specimen testing: the physical removal of a very small sample of the component to perform subsized tests which yield fracture toughness properties or properties which can be correlated to fracture toughness;

b. surface replication testing: microstructural analysis of a surface replica taken from the CSS component of interest, enabling approximation of the fracture toughness by comparing the microstructure of the replica to the microstructure of a body of aged samples whose fracture toughnesses are known.

Subsized Specimen Testing

The advantage of subsized specimen tests is that the actual state of thermal aging/fracture toughness can be determined directly for any component. Disadvantages include the necessity of physical access to the component -- i.e., containment entry and consequent radiation exposure -- and the necessity of gaining regulatory acceptance of the methods.

An ideal materials testing program would take full-size specimens from prolongations (excess material on components), which is clearly not possible in this case. Removal of small boat samples using an appropriate sampling tool (such as that developed at EPRI) will minimize the need for weld repair and makes subsized specimen testing a desirable alternative. An evaluation by Code stress analysis of the small flaw introduced will quickly assess the need for weld repair (14). Careful planning may enable the selection of a sample location at which no repair is necessary.

Table 1 provides a summary of the available subsized specimen test techniques that have been considered for use in a monitoring program. Only a few of the subsized specimen tests offer all the characteristics of sample size and proven method necessary:

-dynamic/static bulge tests
-standard and ball-microhardness tests
-dynamic/static shear punch tests

Microstructural Testing

The obvious advantage of microstructural testing is the nondestructive nature of the test. The disadvantage is that it relies on microstructural model correlations of fracture toughness properties which are not yet fully developed. Because of the progress that has been made towards those correlations and the simplicity and nondestructive nature of the method, however, its development may be desirable. The process requires two types of replicas be taken: a microstructural replica and an extraction replica (20).

Microstructural replication involves making an impression on a small polished and etched surface with a soft polymer material. The replica is then viewed under a light microscope or a scanning electron microscope (SEM) and is used to characterize the amount, morphology, and location of the ferrite phase.

The extraction replica provides carbide size and chemistry. Generally it is made by evaporating a thin film of carbon onto a polished and deeply etched surface. This replica, containing the carbides from the CSS matrix, is then viewed in a SEM or a transmission electron microscope (TEM). A field method of extraction replication is under development by Aptech Engineering Services, Inc.

During inservice inspections, microstructural and extraction replicas could be obtained from CSS components or weldments that were identified as the most susceptible to thermal ageing. Assuming an adequate correlation between microstructure and fracture toughness can be established, the fracture toughness properties of those components could thus be monitored.

Preliminary Small Specimen Testing Results

In support of the feasibilty assessment of small specimen testing techniques, the University of California, Santa Barbara (UCSB), laboratory has performed some small specimen testing of four aged and unaged CF8 samples obtained from Argonne National Laboratory. The four samples were broken Charpy halves from the ANL program whose material properties (strength, toughness) had been determined by full-size sample tests. One sample was unaged; the other three were aged at 400°C for 1000, 3017, and 9998 hours.

The objective of the UCSB testing was to explore the ability of subsized specimen tests to evaluate mechanical property changes. Tests employed were the dynamic and static shear punch and standard 50 gram microhardness tests. Results of the tests are provided in Table 2. Some test results from the ANL program are also included for comparison.

The microhardness test was successful in detecting a significant hardening of the ferrite in the aged CF8 as compared to the unaged CF8. Property changes measured by the shear punch and Rockwell B Hardness tests were not as detectable. This is largely because of the small volume of ferrite and the lack of hardening in the austenite phase. These tests results are only preliminary, however, and the formal proof-of-principle testing will be conducted in Phase One of the Program Plan discussed later.

Fracture surfaces were examined and indicated ductile failure. While dynamic shear punch tests at room temperature also did not indicate a difference, it is believed that transitions in fracture mode could be determined by dynamic shear punch and/or bulge tests at lower temperatures.

Proposed Program Plan

The Program Plan for monitoring primary coolant cast stainless steel components at nuclear power plants consists of three phases whose final objective is the implementation of a surveillance method that is an industry standard with regulatory acceptance. This objective is best achieved with a three phase program in which each phase is designed to achieve the milestones necessary to proceed with the following phase.

The three phases of the Program Plan for monitoring thermal ageing are briefly outlined with their objectives as follows:

Phase One

1) Determine the feasibility of monitoring inservice aging effects by miniature specimen testing and surface replication methods, by showing that these test methods can detect material property changes in materials aged at LWR operating temperatures.
2) Develop preliminary correlations between microstructural and/or micromechanical results with measured fracture toughness changes.
3) Maintain a database of fracture toughness information on aged materials as it relates to structural integrity and long-term degradation.
4) Establish validity of laboratory-induced embrittlement data for predicting material toughness after long-term exposure at LWR operating temperatures.
5) Screen all CSS components at the nuclear plant on the basis of material composition (chemistry) and service temperature. Identify specific components (welds, pipe runs, and RCP casings) that will envelop the embrittlement issue at a plant once incorporated into a surveillance program, thus minimizing the total number of components to be monitored without compromising comprehensiveness.

6) Provide a Position Report on the feasibility and requirements for a surveillance program.

Phase Two

1) Establish the technical basis for the program.
2) Develop procedures for field implementation.
3) Provide technical support documentation for application to ASME Code/NRC Licensing.

Phase Three

1) Implement a pilot plant study.
2) Prepare Pilot Study Report.

Summary

Understanding of the thermal ageing phenomenon in cast austenitic stainless steel has grown rapidly since the mid-1980's, when the five mechanisms which cause embrittlement were identified. Prior understanding was poor and many researchers incorrectly assumed that the embrittlement of materials in the LWR temperature range could be accurately modeled by shorter-term exposure to higher temperatures. Consequently there exists little valid data for the phenomenon as seen in the LWR environment and the full consequence of thermal ageing to the LWR will not be known until more valid tests are completed at Argonne National Laboratory (ANL) in the next few years.

The data that exist clearly show that thermal ageing affects both CSS and weldments by increasing yield and tensile strengths and decreasing fracture toughness. Of the two, the effect on CSS is more dramatic, where in some cases an 80% reduction in fracture toughness is indicated. Even after that reduction, however, there appears to be a substantial reserve of toughness in CSS, and the lowest observed (and published) J_{Ic} for aged CSS remains above that observed in weldments. If, when data for CSS aged at long-term, operating-temperature conditions become available and are published, the toughness of CSS is shown to fall lower than that now assumed for weldments, the question of adequate structural integrity can be posed.

Potentially more significant to the LWR integrity issue is the thermal ageing of weldments. Available test data suggest that the fracture toughness drop observed in aged weldments, while less dramatic than that seen in CSS, can result in a toughness well below the toughness assumed for flux welds in the ASME Code (IWB 3640) evaluation of flaws in austenitic piping. Caution is required in interpreting the data, however, because all available research concerning thermal ageing of weldments is based on accelerated ageing. Clearly, additional research is required.

Given the present conditions -- incomplete research but preliminary indications of a possible integrity issue for aged CSS and aged weldments -- a three phase program for monitoring the thermal embrittlement issue at nuclear power plants has been developed. The thrust of the program is to screen CSS components and weldments on the basis of their susceptibility to ageing (a function of material composition and service temperature) and identify a small but fully representative set for which an active monitoring program is economical and feasible. Such a screening program would identify components that are most susceptible: components with high ferrite, carbon, chromium, and molybdenum contents. The program is phased in order to firmly establish its technical basis for eventual regulatory acceptance. Aspects of the program requiring establishment of a technical basis include the use of subsized specimen or surface replication tests for measuring fracture toughness properties of in-service components.

Acknowledgements

The authors wish to thank Dr. W. Shack of Argonne National Laboratory for making samples of aged CSS material available for small specimen testing. Additional thanks to Dr. G. Lucas of UCSB for his work in the small sample testing of those samples and for his assistance in the preparation of this paper.

TABLE 1. SUBSIZED SPECIMEN TEST SUMMARY

TEST	REFERENCES	SPECIMEN	MEASURES	REMARKS
1. Bulge test (sheet stretching over a hemispherical punch)	12	4.0 cm^2 but has potential for scaling down to TEM disc size	K and n of the parabolic hardening relation, R (planar anisotrop coeff.) and the 3 principle strains	1. Characterizes plastic instability and evaluates a ductile failure parameter
2. Ball microhardness (static indentation tests with spherical indenter)	28	TEM disc size	Chordal diameter of impression repeatedly made by a spherical penetrator	1. $\sigma - \epsilon$ relationship can be determined from the data 2. Sensitive to specimen surface condition; specimen must be prepared and protected before testing 3. Test is sensitive to microstructure; lots of scatter
3. Shear punch	15, 11	TEM disc size	Tensile and yield strengths, work hardening exponent n, reduction in in area in tensile specimens at failure	1. Complements the ball microhardness test 2. Load displacement data can be interpreted in terms of and correlated with uniaxial mechanical properties data 3. The stress and strain fields associated with the process zone are highly complex, interpretation of the test results in terms of analytical models is difficult
4. Small punch	3	10x10x0.5 mm	Load vs. deflection	1. Measures DBTT but gets results lower than those obtained from CVN tests
5. Recrystallization technique and small punch test	19	DCT: 24mm x 5mm thick SP: 10x10x0.4mm	1. Elastic-plastic fracture toughness 2. DBTT	1. Specimens too large.
6. Ultra-low load microindentation nanoindenter	13	Small	Hardness	1. Identifies and characterizes stages of decomposition of the Ferrite Islands 2. Uses mechanical properties microprobe. Microstructural analysis by optical and electron microscopy. 3. Room temp. samples only.
7. Miniaturized disk bend test (MDBT)	2	0.3mm thick, 3mm diameter (TEM specimen)	Ductility	1. Could be used to determine DBTT 2. Much research is required before MDBT can use used regularly, with confidence
8. Subsized compact-tension specimens				
a. J-Integral testing using ASTM procedures	4	Standard C-T geometry w/thickness of 3.8 mm	J-Integral	1. the J_{IC} values obtained in this work were very sensitive to specimen size
b. J-Integral testing based upon microstructural observations	4	C-T's 1mm thick	J-Integral and the stretch zone and the fibrous crack growth	1. The J_{IC} crack initiation value is obtained by plotting the extent of fibrous crack extension, as measured on the SEM versus the J-integral, and extrapolating back to the point of zero fibrous growth
9. In-situ surface replication	20	No restrictions	Microstructure	1. Test consists of two replicas: a. microstructural replica b. extraction replica 2. Expected J_{IC} would be correlated to microstructure

TABLE 2

SMALL SPECIMEN TEST RESULTS AND COMPARISON WITH FULL SIZE SPECIMEN ANL DATA

PROPERTY		VALUE FOR EACH MATERIAL SPECIMEN (Note 1)			
		P12A-02 (unaged)	P13A-03 (400^{o}c/1000h)	P13A-03 (400^{o}c/3017h)	P14A-18 (400^{o}c/9980h)
1. $\Delta\sigma_y$ (ksi)	1mm punch	---	0	12 ± 6	12 ± 4
	3mm punch	---	0	3 ± 3	10 ± 5
	tensile test (Note 2)	---	---	---	5.6
2. ΔUTS(ksi)	1mm punch	---	0 ± 3	0 ± 3	0 ± 3
	3mm punch	---	5 ± 5	6 ± 3	8 ± 3
	tensile test (Note 2)	---	---	---	13.8
3. Rockwell B Hardness		83 ± 2	---	---	90 ± 2
4. Vickers hardness:					
Austenite (50g)		175 ± 10	---	---	170 ± 8
Ferrite (50g)		180 ± 20	---	---	290 ± 26
Combined (500g)		191 ± 7	---	---	205 ± 7
5. J_{IC} (kJ/m^2) (Note 2)		2171 ± 1450	---	---	254 ± 44
6. CVN Impact Energy (J/cm^2) (Note 2)		236.6	67.7	60.7	47.6

Note 1: All test specimens are CF8 material from heat P1 of the Argonne National Laboratory CSS program.

Note 2: Full size specimen test performed by ANL CSS program.

Note 3: Yield and ultimate tensile strength changes are referenced to the unaged material properties.

REFERENCES

1. R.C. Cipolla and E.C. Capener, "Integrity Issues for Cast Austenitic Materials In the Primary Pressure Boundary of Light Water Reactors," EPRI Research Project 2405-16 (draft).

2. M.L. Hamilton and F.H. Huang, "Use of the Disk Bend Test to Assess Irradiation Performance of Structural Alloys," The Use of Small-Scale Specimens for Testing Irradiated Material, ASTM STP 888, W.R. Corwin and G.E. Lucas, Eds., American Society for Testing and Materials, Philadelphia, 1986, pp. 5-16.

3. J.-M. Baik, J. Kameda, and O. Buck, "Development of Small Punch Tests for Brittle-Ductile Transition Temperature Measurement of Temper-Embrittled Ni-Cr Steels," The Use of Small-Scale Specimens for Testing Irradiated Material, ASTM STP 888, W.R. Corwin and G.E. Lucas, Eds., American Society for Testing and Materials, Philadelphia, 1986, pp. 92-111.

4. P. McConnell et al., "Experience in Subsized Specimen Testing," The Use of Small-Scale Specimens for Testing Irradiated Material, ASTM, STP 888, Philadelphia, 1986, pp. 353-368.

5. O.K. Chopra and H.M. Chung, "Effect of Low-Temperature Aging On the Mechanical Properties Of Cast Stainless Steels," Properties of Stainless Steels in Elevated Temperature Service, MPC-Vol.26/PVP-Vol.132, M. Prager, Ed., American Society of Mechanical Engineers, New York.

6. G. Slama et al., "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Castings and Welds," SMIRT Postconference Seminar 6, Monterey, California, 1983.

7. E.I.Landerman and W.H. Bamford, "Fracture Toughness and Fatigue Characteristics of Centrifugally Cast Type 316 Stainless Steel Pipe After Simulated Thermal Service Conditions," ASME MPC-8, 1978.

8. O.K. Chopra and H.M. Chung, Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems: Semiannual Report, October 1986 - March 1987, NUREG/CR-4744 Vol.2, No.1, ANL-87-45, July 1987.

9. Evaluation of Flaws in Austenitic Steel Piping, EPRI NP-4690-SR, Special Report, July 1986, Electric Power Research Institute, Palo Alto, CA.

10. P. McConnell and B. Sheckherd, Fracture Toughness Characterization of Thermally Embrittled Cast Duplex Stainless Steel, EPRI NP-5439, Final Report, September 1987, EPRI, Palo Alto, CA.

11. F.M. Haggag et al., "The Use of Miniaturized Tests to Predict Flow Properties and Estimate Fracture Toughness in Deformed Steel Plates," presented at the International Conference on Fatigue, Corrosion Cracking, Fracture Mechanics and Fatigue Analysis, Salt Lake City, Utah, December 1985.

12. M. Dooley, G.E. Lucas, and J.W. Sheckherd, "Small Scale Ductility Tests," J. Nuclear Materials, 103 & 104, 1981, pp. 1533-1538.

13. S.A. David et al., "Nanoindentation Microhardness Study of Low-Temperature Ferrite Decomposition in Austenitic Stainless Steel Welds," Welding Research Suppl., August 1987, pp. 235-s to 240-s.

14. J.F. Copeland et al., "Field Certification of Piping Material," EPRI Project RP-2680-1, Draft Report, June 1987, Electric Power Research Institute, Palo Alto, CA.

15. G.E. Lucas et al., "Shear Punch Tests For Mechanical Property Measurements In TEM Disc-Sized Specimens," J. Nuclear Materials, 122 & 123, 1984, pp. 429-434.

16. H.M. Chung and O.K. Chopra, "Kinetics and Mechanism of Thermal Aging Embrittlement of Duplex Stainless Steels," October 1987, to be published in the Proceedings of the Third International Symposium On Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 30 - September 3, 1987, Traverse City, MI.

17. H.M. Chung and O.K. Chopra, "Characterization of Duplex Stainless Steels by TEM, SANS, and APFIM Techniques," June 1987, presented at the International Metallographic Society Symposium on Characterization of Advanced Materials, July 27 and 28, 1987, Monterey, CA.

18. G.E. Hale, J.R. Jordan, and B. Hemsworth, "Fabrication and Ageing of Cast Austenitic Steels," to be published in the Proceedings of the Third International Symposium On Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 30 - September 3, 1987, Traverse City, MI.

19. T. Misawa et al., "Small Specimen Fracture Toughness of HT-9 Steel Irradiated With Protons," J. Nuclear Materials, 133 & 134, 1985, pp. 313-316.

20. "Field Usable Extraction Replication of Duplex Stainless Steels for Monitoring Service-Induced Material Degradation," Aptech Engineering Services, Inc., proposal to SBIR.

21. A. Vignes, "Understanding the Phenomena of Materials Degradation By Aging and Embrittlement - Margin of Safety and Solutions to Limit or Avoid Such Degradations," Framatome, France (date unknown).

22. F.R. Drahos, W.L. Server, and B.F. Beaudoin, "Light Water Reactor Coolant Pumps," Residual Life Assessment of Major Light Water Reactor Components - Overview, Volume 2 (Draft), NUREG/CR-4731, EGG-2469 Volume 2, March 1988, V.N. Shah and P.E. MacDonald, Eds., EG&G, Idaho Falls, Idaho.

23. M.K. Miller et al., "Long Term Thermal Aging of Type CF 8 Stainless Steel," Journal De Physique, Colloque C9, supplement au no. 12, Tome 45, Decembre 1984, page C9-385.

24. J. Bentley and M.K. Miller, "Combined Atom-Probe and Electron Microscopy Characterization of Fine Scale Structures in Aged Primary Coolant Pipe Stainless Steel," Material Resource Society Symposium Proc., Vol.82, 1987.

25. D.T. Umino and A.K. Rao, Long-Term Inspection Requirements for PWR Pump Casings, EPRI NP-3491, Final Report, May 1984, Electric Power Research Institute, Palo Alto, CA.

26. A.J. Giannuzzi, LWR Experience With Centrifugally Cast Stainless Steel Pipe, EPRI NP-4996-LD, Final Report, December 1986, Electric Power Research Institute, Palo Alto, CA.

27. Telecon between T. Hardin (Robert L. Cloud & Associates, Inc.) and O. Chopra (Argonne National Laboratory), June 16, 1988.

28. G.E. Lucas and N.F. Panayotou, "Microhardness Tests for High Energy Neutron Source Experiments," J. Nuclear Materials, 103 &104, 1981, pp. 1527-1532.

EFFECT OF AGING ON PERFORMANCE OF NUCLEAR PLANT RTDs

H. M. Hashemian
K. M. Petersen

ABSTRACT

The accuracy and response time of resistance temperature detectors (RTDs) can suffer degradation with aging. The rate of degradation depends on the quality of the RTD, its installation, and the process conditions. This paper presents some examples of performance problems due to aging and other factors in nuclear power plant RTDs.

INTRODUCTION

A number of important safety related temperature measurements in nuclear power plants are made with RTDs. These sensors must therefore be accurate and have a fast response.

The accuracy and response time of RTDs are generally independent and are therefore treated separately. The accuracy is a measure of how well an RTD provides steady state temperature information, and the response time is an index for determining how fast it can reveal a change in temperature. The accuracy is determined from the calibration of the RTD and can be improved by recalibration. The response time, on the other hand, is an intrinsic characteristic of the RTD which cannot be altered without mechanical repositioning of the RTD constituents.

Experience has shown that both the accuracy and the response time of nuclear plant RTDs can suffer degradation by aging. Therefore, periodic calibration and response time testing are performed to ensure that acceptable performance limits are maintained while the plant is operating. Fortunately, a simple method is available for response time testing of RTDs while they remain installed in an operating plant. The method is called the Loop Current Step Response (LCSR) test. The LCSR test was developed specifically for in-situ response time testing of nuclear plant RTDs[1] This test is used by many utilities worldwide to verify that the response times of safety system RTDs are acceptable.

Unfortunately, a practical in-situ method is not available for accurate calibration of RTDs. Presently, the only correct way to determine the accuracy of an RTD is to perform laboratory calibration which requires removal of the RTD from the plant. This approach is used by very few utilities in the country. This is because removal of RTDs is inconvenient and can cause damage to the RTD. Therefore, the accuracy of RTDs is often not verified except by cross-calibration which is usually performed once after each refueling outage. The cross-calibration approach as currently used can reveal gross accuracy problems but is not effective when systematic or unidirectional drift is involved.

As there had been no systematic study on the rate of calibration shift in nuclear plant RTDs, a project was initiated in the Fall of 1986 to quantify the effect of aging on performance of RTDs[2] To date, the project has indicated that drift induced by aging in RTDs may deserve attention. This project is sponsored by the Nuclear Regulatory Commission (NRC) and will be continued for at least one more year. The key purpose of this project is to identify the achievable accuracy limits for nuclear grade RTDs and quantify the effect of thermal aging on calibration. It is anticipated that this project will provide objective information to help in establishing optimum testing or replacement schedules for nuclear plant RTDs.

BACKGROUND

Two Regulatory Guides issued by the NRC in the mid 1970s initiated the interest in sensor response time testing and sensor calibration in the nuclear industry. These are Regulatory Guide 1.118 entitled "Periodic Testing of Electric Power and Protection Systems," and Regulatory Guide 1.105 entitled "Instrument Setpoints."

The NRC specified in these documents that means shall be provided to permit periodic performance verification of safety related instrument channels including the sensors. It was recommended that the sensors be tested while installed in the plant, when possible, or otherwise be removed and tested in a laboratory. These recommendations were intended to include RTDs and pressure sensors. For testing of RTDs, research began very soon after the guides were issued to find a testing method that could be performed remotely while the plant is at normal operating conditions (in-situ testing). The work concentrated on adaptation of the Loop Current Step Response (LCSR) method for response time testing and the Johnson noise technique for calibration. The implementation of the LCSR method in nuclear power plants did not encounter major difficulties as did the Johnson noise technique. As a result, response time testing is routinely performed at normal operating conditions in many nuclear plants, but in-situ calibration is still a problem. This is because the accuracy of the Johnson noise technique is dependant on the length of the cables between the RTD and the calibration equipment[3] A significant amount of research has been completed at the Oak Ridge National Laboratory (ORNL) to resolve the cable problem, but more work is needed before the Johnson noise technique can be used effectively for remote calibration checks of RTDs at plant operating conditions.

Presently, the only means of in-situ calibration for nuclear plant RTDs is the cross-calibration method. A recent informal examination of the cross-calibration practice and procedures in nuclear power plants indicates that the effort involved in implementation of this approach is reasonable and plant personnel are reasonably satisfied with its effectiveness. However, the test procedures can be improved and computer-aided scanning with signal validation techniques can be incorporated to minimize some of the inaccuracies in the cross-calibration approach. In a few plants, thermocouples are sometimes included in obtaining the average process temperature for use in RTD cross-calibration. This is not always advantageous because thermocouples are generally not as accurate as RTDs.

RTD PERFORMANCE REQUIREMENTS

The requirements for RTD response time and accuracy vary widely in the nuclear power industry. The response time requirements are usually specified in terms of a time constant which is the conventional index for quantifying the dynamic characteristics of temperature sensors.

The accuracy requirements, however, are not specified in terms of a unique index. This creates confusion in comparative evaluation of sensors and in verifying the adequacy of RTDs for providing accurate temperature information. This problem is not unique to the nuclear industry. It stems from the lack of a useful standard for unambiguous quantification of RTD accuracy.

The current response time requirements are generally reasonable throughout the nuclear industry. However, the accuracy requirements vary by an order of magnitude from practically unachievable to the opposite extreme. For example, in a pressurized water reactor, the primary coolant RTDs are required to provide accuracies of better than three tenths of a degree Fahrenheit in measuring hot leg temperatures of about 600°F. Such requirements are difficult to verify by conventional cross-calibration. The RTDs must be removed from the plant and calibrated in a laboratory to prove that they can measure temperatures of about 600°F to within a few tenths of a degree. Although such accuracy requirements can be achieved by laboratory calibration, it is difficult to prove that they can be maintained for more than a fuel cycle at plant operating conditions.

AGING EFFECTS

The aging effects on RTD performance may be separated into two groups: aging effects on response time; and aging effects on calibration or accuracy. The aging effects on response time are adequately understood in the nuclear industry, but aging effects on calibration are still unknown.

The aging effects on response time are known because response time testing has been performed in many plants for more than ten years. These have revealed many problems, a majority of which have been resolved. Three of the worst examples of RTD response time problems in nuclear power plants are shown in Table 1. These are from three different pressurized water reactors. The results are given in terms of the response time required for the RTD versus the actual response time measured by in-situ testing at or near normal operating conditions.[4] Note that in one of the three cases, the response time of the RTD was more than six times larger than allowed by that plant's technical specifications. Of course, the plant was shutdown when the problem was discovered. The cause was identified as insufficient insertion of the RTD into its thermowell. It is important to point out that the response time of a well-type RTD is very sensitive to the fit between the RTD and its thermowell at the sensitive tip. It has been shown[4] that a small axial or radial gap at the RTD/thermowell interface can cause a significant change in response time.

TABLE 1

Worst Examples of RTD Response Time Problems

Time Constant (Sec.)	
Expected	Actual Value
6	37
5	21
4	12

Above results are from in-situ tests at three operating plants.

Table 2 shows how aging affected the response times of a selected number of RTDs after an 18 month operating cycle. There are also examples of aging that show improvement in response times. These are given in Table 3. It must be mentioned that the results in Table 1 and 2 are representatives of problems encountered in the past and are not typical for the current RTD installations in nuclear power plants. Nevertheless, problems are still encountered with RTD response time, but their frequency and magnitude are less than those experienced before 1985.

TABLE 2

Worst Examples of RTD Response Time Degradation in One Operating Cycle

Time Constant (Sec.)	
Initially	18 Months Later
5.6	12.0
4.1	10.5
2.7	4.5

Above examples are from in-situ tests at normal operating conditions.

TABLE 3

Examples of Improvements in Response Time with Aging

Time Constant (Sec.)	
Initially	18 Months Later
6.2	5.0
6.7	5.7
2.7	1.8

Above examples are from tests in nuclear power plants at normal operating conditions.

Some of the response time degradation problems with the well-type RTDs are due to degradation of a thermal compound called Never-Seez which has been used in some RTD thermowells for response time enhancement. This compound is normally used as an industrial lubricant in mechanical equipment. In its fresh state, Never-Seez is effective in reducing the sensor response time (Table 4). However, the compound often degrades at reactor operating temperatures and loses its useful heat transfer properties. As a result, it can become more like an insulator than a conductor. Although improved thermal couplants are now available, their use for response time improvements in RTDs has become unpopular in nuclear plants due to a history of degradation problems.

A new approach for response time enhancement is to electroplate the sensing tip of the existing RTDs with silver or gold in order to fill the gap between the sensor and the thermowell. This has been done in some plants with positive results. However, the approach is new and adequate data is not available to determine if this approach can be viewed as a long term solution. There are some questions about the plating approach. These questions stem from concerns about the seizing of plated RTDs in thermowells, neutron activation of gold, silver oxidation and swelling, etc. In addition, plating of the sensor to eliminate the air gap at the sensor/thermowell interface may cause a shift in RTD calibration. This can occur if the sensor is tightly squeezed by the thermowell during installation or because of thermal expansions of sensor materials at elevated temperatures. The sensing element at the tip of the sensor is extremely sensitive to stress. Stress can cause the element to open or shift in resistance. RTDs which have been tightly installed in thermowells are more likely to fail or experience calibration drift than those that have been installed in a way that permits some free movement with respect to the thermowell. RTDs have been found to suffer open circuits when they were forced-fit into thermowells. It is also important to mention that the plating process itself may cause shifts in RTD calibration. Therefore, the plating of RTDs must be done with utmost care in order to avoid any stressing of the sensing tip. It is apparent that there may be a trade-off between enhanced response time and long-term reliability of plated RTDs.

TABLE 4

Effect of Never-Seez on RTD
Response Time

Sensor No.	Time Constant (Sec.) Without Never-Seez	 With Never-Seez
1	6.5	4.2
2	5.3	4.3
3	5.3	3.9
4	8.0	5.7
5	14.8	8.1

Above results are from laboratory tests in room temperature water flowing at 3 feet per second.

Some examples of RTD calibration problems are given in Table 5. Note that one RTD shifted in calibration during storage. This is believed to be due to moisture ingress through the RTD seal. The remaining examples are from experience at plant operating conditions. It must be pointed out that these are not typical examples, they are the worst examples that the authors are aware of with respect to qualified RTDs in the safety systems of nuclear power plants.

TABLE 5

Worst Examples of RTD
Calibration Problems

Temperature Error	Description
8°F	Calibration shift in two years with the RTD in storage
5°F	Discrepancy between two RTDs reading the same temperature
1°F	Error due to thermoelectric effects
5°F	Difference between two elements of a dual RTD
1°F	Improper wire shielding
5°F	Drift in Delta T across the core

RTD AGING RESEARCH

A research project sponsored by the NRC is currently underway to quantify the effect of aging on RTD calibration. The project involves thermal aging of a number of RTDs representing a few from each of the common manufacturers of nuclear grade RTDs. The RTDs are periodically removed from the aging process and recalibrated. During the aging process, the RTDs are scanned and their insulation resistance, loop resistance, and any thermoelectric voltage are recorded. Early results have not provided a unique rate for calibration drift of these RTDs but have shown errors of up to 2°F in a few RTDs in less than one year of aging. The results also indicate that a significant portion of RTD drifts occur early in life. That is, the RTD drift stabilizes after a few months of exposure to normal operating temperatures. The latter conclusion is important because it suggests that RTDs may remain more stable if they are heat treated before they are installed in a plant.

The project mentioned above is the first systematic study of aging effects on calibration of industrial RTDs. The only other significant contribution in this area is from work at the National Bureau of Standards[5]

CONCLUSIONS

Aging effects on the performance of safety system RTDs can be significant and should be accounted for by periodical response time testing and calibration. This can be done with test methods which are available and can be implemented on installed RTDs while the plant is operating.

To account for response time degradation of RTDs, most utilities are using an in-situ method called the Loop Current Step Response test. The tests in most plants are performed once every refueling outage which is a suitable frequency for those plants with a better than 20 percent margin between the nominal response time values and the required values.

The degradation of RTD's accuracy is presently accounted for by cross- calibration. Although improvements can be made in cross-calibration procedures, the method is adequate for detecting gross calibration shifts. A research sponsored by the NRC is underway to establish objective test frequencies for calibration of nuclear power plant RTDs. The final results will be presented in a NUREG/CR report to be published in mid 1989. Early results indicate that calibration shifts occur in RTDs and suggest that cross-calibration performed after each refueling outage is a reasonable measure to detect the outliers. However, to measure high temperatures (300 to 700° F) with accuracies of better than a degree, each RTD must be calibrated in a laboratory.

REFERENCES

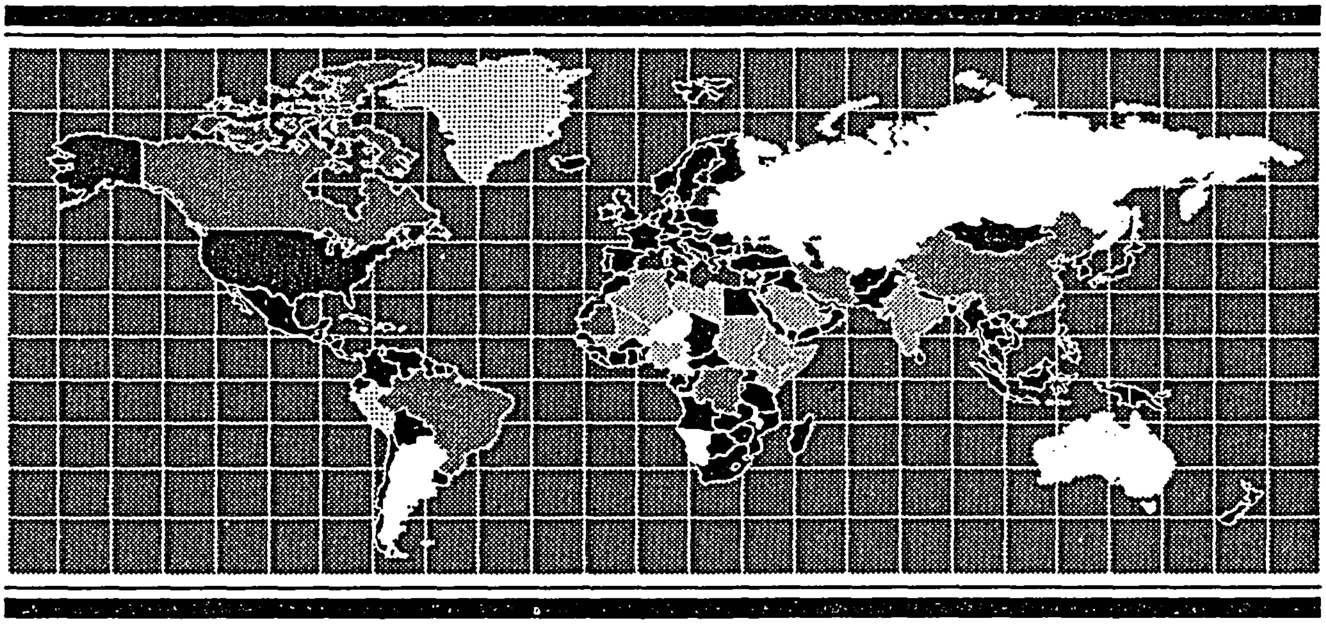
1. Kerlin, T. W., et al., "Temperature Sensor Response Characterization," Report No. NP-1486, Electric Power Research Institute, Palo Alto, CA, August 1980.

2. Hashemian, H. M., et al., "Degradation of Nuclear Plant Temperature Sensors," NUREG/CR-4928, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1987.

3. Blalock, T. V., "Remote Calibration of Resistance Temperature Devices (RTDs)," Report No. NP-5537, Electric Power Research Institute, Palo Alto, CA, February 1988.

4. Hashemian, H. M., Petersen, K. M., "Calibration and Response Time Testing of Industrial RTDs," Proceedings of the 34th International Instrumentation Symposium, Test Measurement Division of the Instrument Society of America, Albuquerque, New Mexico, May 1988.

5. Magnum, B. W., "Stability of Small Industrial Platinum Resistance Thermometers," Journal of Research of the National Bureau of Standards, Vol. 89, No. 4, pp 305-316, July-August, 1984.



PANEL SESSION

September 1, 1988

Session Chairman

PROF. DR. LEONARD V. KONSTANTINOV
Deputy Director General
International Atomic Energy Agency
Austria

Session Co-Chairman

GUY A. ARLOTTO
Director, Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commisson

PANEL SESSION : Opening Statement by the Chairman

Prof. Leonard V. Konstantinov

Nuclear power has proved its maturity, producing annually more than 16% of world electricity. In fact, in a few countries it produces more than 40-50%. In accordance with IAEA forecast, nuclear power will double its output by 2000.

We know well that nuclear power saved people some time ago from severe cold in winter, and awful heat in summer, but accidents like TMI and Chernobyl constituted a drastic setback for public acceptance. We know that there have been no new orders for nuclear power plants in the U.S.A. since 1978. Some countries have cancelled their intention to develop nuclear power, and some even plan to close their nuclear power plants inspite of their good performance.

But what is the real alternative to nuclear power to maintain the standard of living in industrialized countries and increase it in developing countries? Scientists and engineers see only burning of fossil fuel and hydro - where it has not already been used. The demand for electricity is steadily increasing, in parallel - if not faster - than the growth of GDP, as, i.e. in the U.S.A. at present 37% of primary resources are used for electricity production. There is no serious change anticipated in this trend to increase the demand for electricity in the future.

Consequences of burning fossil fuel for the environment and mankind are well known by now, like acid rain, dying forests, and converting lakes into salt marshes. Further on, scientists forecast serious consequences from the greenhouse effect; and the problem of environmental pollution worries people, including the Green Peace movement, more and more. In these circumstances proposals to extend the life of nuclear power plants is one of the ways to save nature.

Many nuclear power plants built more than two decades ago have given good economic service with good safety records. Some of them have been working very well even beyond their originally planned design life. By the close of this century approximately 160 plants will be 25 years old, and about 70 will be 30 years old or more. As to the year 2010 more than 300 plants will overrun the 25-year mark. With growing experience and with even better technical performance of the plants, the urge to keep them running beyond their planned design life will be stronger. The economic gains accruing to society from postponed capital investments for replacement plants and from obtaining power from plants whose costs have been fully amortised, will be very large.

However, all the gains are possible only if the plants are run very safely meeting the safety standards expected of them. We know that ageing affects the expected performance of reactor structures, systems and components. It could result in increased wear out, potential common cause failures, and general reduction in plant life. The extent of ageing also depends upon the environment in which the plant has been working, the manner of operation and maintenance, the quality of the original equipment, and quality of construction. The quality level of a plant at any point of time in its operating life also depends upon the repair and replacement of components and systems which have gone beyond their desired performance requirements. Thus an ageing plant is a complex combination of many aged components and systems which have aged to different extents in terms of performance and also in terms of time under usage. This state is also continuously changing, and is plant specific, even for similar plants.

Safety Aspects

The problem of evaluating and ensuring safety for new plants to be licensed is an arduous time consuming process. The problem of doing the same for an aged plant would be more difficult, considering the fact that one would have to go through the ageing history of each of the safety related equipment and systems. To some extent, we have been going through the process all the time when the safety of all operating plants is evaluated and the plants are permitted to continue operation. But I can foresee that the safety authorities and the public would, in course of time, demand a more formal methodology for assuring safety for plants which are nearing the end of their design life. Besides the plants were designed and accepted for operation meeting the safety standards that were prevalent at the time of initial licensing, while over a period of time the requirements of safety standards would have become more stringent. The problem of backfitting and allowing older plants to continue to operate under current safety requirements is a difficult one and has also to be addressed.

The incentive for continuing to operate an aged but otherwise healthy plant is very great, and all of us here are keen to understand the problem clearly and to try to find solutions for it.

Understanding Ageing

The general phenomenon of ageing has been a much discussed topic but attempts to scientifically and comprehensively understand this complex phenomenon for NPPs are of more recent origin. The difficulties in studying the subject are aggravated by the inability to realistically simulate the ageing environment in the laboratory. In addition, the process of ageing is slow in comparison to the time span in which one would desire to observe and obtain data. Again the environment in which ageing takes place in the field is multitudinous and not easy to list.

The only manner in which we would be able to effectively proceed to understand ageing is to go through as many historical records of plants and equipment as have been kept in all countries, in addition to carrying out laboratory tests under simulated conditions. Therefore, the first requirement is the establishment of a data bank from information available in all the countries. Next the step of analysing and understanding the information hidden in the data bank will have to be undertaken. Planning of laboratory investigations should proceed side by side. Not to be forgotten are the post-service examinations of components and systems from running plants in order to extract ageing related information and regular field assessments for noting ageing degradation.Improved instrumental and analytical techniques may have to be developed.

In short, I believe that a whole new branch of science and technology on ageing will be developed in the near future to resolve the problem of ageing and to extend the life time of nuclear power plants whenever feasible.

International Co-operation

From what has been briefly stated above, the enormity of the task would have already been realised. It would also be very expensive. The only way out would be strong international co-operation for the job. As already mentioned, data are important for using physical and statistical methods of analysis, and such data are slow in coming by. Enough data and data under varied conditions of ageing can only be obtained from a large number of plants spread all over the world. Hence, apart from costs, international co-operation is imperative from the technical view point also.

In view of the shortage of resources everywhere, the tasks have to be worked out selectively, concentrating mostly on the critical components and systems. Naturally, items critical to safety will be of importance, but, as the questions of safety will arise only if there is capability of running the whole plant beyond the original design (or financial) life, the reliability of other critical areas of the plants which have aged will also have to be considered.

From all the studies will evolve the methodology for mitigating the effects of ageing. Not only will we be able to assess the residual life of components, but we would be in a position to note incipient failures because of ageing. It will facilitate the replacement of components before failure, but not too early - which would again be uneconomic. One could expect not only a higher degree of safety but also higher capacity factors from plants. It would also lead to better preventive, predictive and corrective maintenance; better refurbishments and replacements; and better equipment design, manufacture and qualification methods. I am looking forward to the initiation of closer international co-operative effort for this whole process.

The I.A.E.A. has initiated a programme in this area, and will be happy to offer its services, within the limitation of its resources, to this international activity. International working groups on reliability of reactor pressure components and on control and instrumentation advise the Agency in this field to organize international meetings, similar to the International Symposium on Safety Aspects of the Ageing and Maintenance of Nuclear Power Plants which took place last year in Vienna, and to provide other fora for information exchange. The Agency closely co-operates with other international organizations to promote international co-operation in the area of NPP ageing.

COMMENTS

by

Guy A. Arlotto

U.S. Nuclear Regulatory Commission

I thought that, since several of the panel members were not with us for the last 2½ days, I would set the stage by giving them a thumbnail sketch of my impressions of what has transpired.

I would like to say that we have heard many excellent technical papers. There is a great deal of work going on. Most of us will agree that we will have to get the proceedings to really sort out the technical information presented here.

There was agreement that aging can affect safety. The principal concerns from a safety viewpoint were the potential for common mode failures and for reduction of defense in depth, which has been our basis for reactor safety for many years. The principal reason for pursuing nuclear plant aging research is to turn off potential aging-related accidents before they occur.

I got the impression there were several things we could do to better understand and manage aging. The first is that we can use experience from other industries. And, at this conference, we have heard reference to the United States Navy experience which the speaker thought was analogous to some of our problems; we also heard references to airline industry experience. So I think that other industries' experience is one area we should pursue.

Accumulating and recording operating experience accurately and carefully is a second thing that was said over and over again. The third thing was to record repair and replacement. These data are very important if we are to understand and manage the aging problem.

The sharing of information was a key theme in just about every paper that was given. Simply the fact that people were giving this information was an indication they thought it was extremely important to share.

Going further though, there were suggestions that possibly we could set up a centralized data system, maybe worldwide, in order to accumulate some of these data that I just outlined.

Also identified was a need to build on and upgrade codes and standards to account for aging. Of course, there are certain professional societies in the United States that have already begun to undertake this, namely the ASME and IEEE. The ASME has a whole committee called Operation and Maintenance. In addition, you have heard discussion of Section XI of the ASME Boiler and Pressure Vessel Code which is principally directed at in-service inspection.

Research on aged equipment is another key element by which we can attack this problem. It was pointed out that we cannot let targets of opportunity go by, because once a particular component that has been aged naturally is decommissioned or removed from service we must get those things we need quickly, or they will be buried and be lost forever.

Another very important thing I would like to mention is the need to develop detection techniques that can produce reliable results we can use in decision making regarding safety. In particular, nonintrusive techniques would be very useful to identify if there were an intrusion into a system. This was brought up by a speaker from the General Electric Company, who was concerned that we may end up with a corrosion problem simply because some raw water had gotten into a system. If we do not have some kind of monitoring system, we would not know whether observed corrosion was from natural aging, or if it was from this special situation.

In-service inspection and testing are areas we are working on very hard, and we have come a long way. I do think we need to do better, particularly in the area of non-metals in-service inspection so we can test cables or other non-metallic components that are important to safety. Also, we must be capable of detecting the onset of deterioration for active components, which would go a long way in giving us trending data before too much degradation has occurred.

There are many related subjects that we are grappling with in which aging research could have an input. Maintenance comes to mind immediately, where the results could be utilized so that we can better identify what to look at, how to look at it, and how often.

Frequent mentions were made of license renewal and life extension throughout this meeting. This should be an area where aging data would be useful to trend degradation, so we could really determine what residual life is left. I am very concerned about an approach that says pull out and replace everything whether you need to or not. I think that we have to be very careful that neither we as the regulators nor the industry take that approach.

Yet another related area is in equipment qualification where aging data would be very useful as a basis for more accurately simulating artificial or accelerated aging, which is clearly the only way to really qualify a piece of equipment for its entire life.

The next item was the inclusion of aging data into PRA models that, based on real data, would give us more realistic credible risk insights. In addition, I would say that in the area of PRA it cuts both ways, since PRA itself could help us identify those items for aging research of higher priority from a risk point of view. I think this was pointed out this morning.

And lastly, the results of our aging research could be useful in identifying where diagnostic and monitoring methods need to be incorporated into the design for standardized or future plants for assuring adequate accessibility, particularly for performing ISI, testing, and surveillance.

In closing, I am just going to say that we would be interested in hearing panelists' thoughts on what is going on in aging and managing aging, and what each of them believes is the proper role of this segment of nuclear activity and where they think it could make the biggest contribution.

PLANT AGING AND LIFE EXTENSION -- A UTILITY PERSPECTIVE

Wallace B. Behnke, Jr.

At Commonwealth Edison, we have been managing our way through the nuclear power plant aging process since Dresden 1 went in service in 1960. We were not alone in this experience. Worldwide, a great deal of experience has been accumulated to date. Industry and the NRC are vigorously addressing the unresolved issues. The professional societies have been reflecting the growing body of knowledge in our system of codes and standards. NRC is to be commended for convening this symposium as a focal point for assessing current knowledge and setting the stage for development of an agenda for the future.

As I see it, there are two objectives to be pursued. First, improving the operation of existing facilities and second, the possibility of life extension. With respect to the first objective, current industry practices are generally adequate to assure safe and reliable operation over the initial 40 year license term. With respect to the second objective, let me summarize my views with six points.

1. Life extension is an attractive possibility and a worthwhile objective.

2. Improved understanding of the technical aspects and economic considerations involved in extending the service life of certain materials, equipment, and structures is a necessary prerequisite to selection by utilities of a life extension strategy.

3. While there is nothing magic about the 41st year of plant life, there is a need to develop an R&D agenda to fill in the gaps in current knowledge. The R&D should concentrate on increasing the understanding of the life expectancy of critical components (important to safety) not normally replaced or refurbished during the initial licensing term. It is important to recognize that:

   -- Certain plant life extension initiatives may have to be undertaken early if maximum economic benefits are to be fully realized;

   -- Even if life extension doesn't pan out, the money focused on R&D will be of value because the information will be useful to continuing operations as well as for the design of the next generation of nuclear power plants; and

   -- Efforts to address the issues must not be allowed to undermine public confidence.

4. NRC guidelines for the relicensing process are also urgently needed. Important questions remain to be answered, i.e.,

   -- What will be the period of relicensing?

   -- To what extent will older plants be grandfathered against future regulatory standards of increasing severity?

   -- What criteria must be satisfied for relicensing?

5. Codes and standards will need to be reviewed with life extension in mind, and this work is already well along and being coordinated by ASME. The technical community seems to be on top of the issue.

6. In the final analysis, selection by utilities of a life extension strategy will be determined by site specific economic analysis of alternatives using present value analysis of revenue requirements methodology and taking into account financial risks and other factors. Such a strategy should recognize that the extend vs. replace decision must accommodate the lead time for new capacity.

Conclusion

There is still a lot to be accomplished before it will be safe to bet the integrity of the nation's power supply on a life extension strategy, especially given the long lead times required for replacement capacity.

DELBERT F. BUNCH
U.S. Department of Energy

Much has been said at this conference about the importance of understanding and managing plant aging. There are a few general observations I would make in the hopes of putting these discussions in context.

First:
We are not sailing in uncharted waters. Aging phenomena are generally well understood. New problems will arise less frequently as our experience base grows. Safety requirements may cause some plants to be derated or close prematurely, but many more plants will be candidates for operation well beyond 40 years.

Second:
There are undoubtedly some special technical issues associated with aging power plants that have yet to be fully analyzed, but fundamentally the underlying issue is how long equipment will last that cannot be economically replaced.

Third:
Regulatory requirements will vary from country to county, but decisions on life extension will turn on economic issues, and not safety and not broader national or international security interests.

The operating reactors in this country are an important national resource. They represent an investment of over $200 B. They produce nearly 100 GWe. Worldwide, nuclear power replaces the equivalent of 8 mbbl/d of oil. We should all be committed to using this resource in a safe and efficient manner and as long as possible.

The capacity factors of U.S. reactors are, on average, 10-20% too low. A rise of 1 percent of average capacity factor is equivalent to putting a new 1000 GWe reactor on the grid. Failure to deal with aging properly will impact plant availability.

But economics aside, plants that experience problems fuel public concerns about the safety and economics of nuclear power. Problem plagued plants will not be encouraged to keep on running by the public, by the rate commissions or by the NRC - even if the plants produce at reasonable rates.

Finally, a whole new generation of advanced reactors are available to meet the need for expanded capacity but unless we can show we can operate today's plants in a safe, economical manner, there will be a limited future for nuclear power in this country.

In this context I have some reservations about over-emphasizing the scientific aspects of aging. We do need to understand how plants age, but the fact is that basic attention to proper care of a power plant has not been uniformly practiced. Since different operators are getting both very good and very bad performance out of similar reactors, it is the method of management of these plants that needs to be the focus of our attention.

O/M costs for nuclear plants continue to rise. FERC recently reported an average 33 percent increase in the 3-year period from 1985 to 1987. Much of these costs seem to be the result of regulatory actions that have little to do with assuring our plants are well managed. Misdirected attempts to address plant aging as a strictly technological issue can become another contributor to this trend. Care must be taken not to make this another long term research program which goes on and on, leaves industry in a continued state of uncertainty, sours relations between NRC and industry and NRC and Congress, and in the end increases cost but doesn't really benefit safety or increase a practical understanding of the life time capabilities of plants.

The regulators need to work constructively and cooperatively with the Nation's utilities to learn from their successes and not to prescribe a set of regulations which, while well-intentioned, penalize the good performers and effectively prevent utilities from investing in life extension.

The national and international security benefits from life extension require that we approach this issue with a determination to find sensible solutions to what is fundamentally a straightforward issue.

A. Bert Davis
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission

Thank you, Mr. Chairman, and good afternoon ladies and gentlemen.

I am pleased to be with you today as a Panel Member of this International Nuclear Power Plant Aging Symposium. I understand that we have been designated as expert panel members. Apparently that would signify that I am an expert in managing aging and life extension of nuclear power plants. To the contrary I am a Nuclear Engineer generalist and the Administrator of a Nuclear Regulatory Commission (NRC) Regional Office. My past experience does include the studies of behavior of materials and components during and after in-pile radiation effects experiments, and it includes plant construction, testing, and operation as well as regulation. Therefore, my comments today come from the perspectives of understanding the complexity of designing and performing experiments to obtain meaningful data, and of understanding that after completing construction, it is a significant challenge to operate a nuclear power plant well and in a manner which properly accommodates aging.

Life extension beyond forty years, and for that matter managing aging to safely operate a nuclear power plant for up to forty years, is a challenging, sobering and necessary undertaking in my view.

We are dealing with nuclear power plants and their inherent risks. In spite of these risks, the safety record that we have achieved to date in the United States has been quite impressive. To assure that we continue this safety record, however, we must take proper actions regarding the aging of these plants. We must assure that the aging processes do not result in adverse effect on public health and safety.

We are dealing with a very complex process in understanding the various aging mechanisms and their synergisms. It involves the interactions between the materials used in the components, systems and structures and the environment in which they operate. In my view, understanding and dealing with this complex process is more difficult than what we have faced so far in the design, construction, testing and operation of these plants. Yet major NRC Safety System inspections have identified significant management, design, procedure, modification, maintenance and record keeping deficiencies. Perhaps more importantly we continue to find, as a result of failures or events, new mechanisms which were not predicted or expected. And such mechanisms are probably easier to predict than many we will face as the plants age.

Going back to the sixties, we have had research programs to better understand some of the aging phenomena such as irradiation-induced embrittlement, fatigue, corrosion, erosion, and chemical attack. This information should help us to understand and manage aging. In my view it is important that we aggressively pursue the job of managing aging now rather than rely on an evolving process as we often experienced in managing design, construction, testing, and operation in the past. This will be necessary to retain our excellent safety record.

We are dealing with strong economic incentives to extend the life of a major investment. We must not let this factor override a careful, deliberate approach to determine the viability of life extension and to properly manage aging.

My remarks so far may appear somewhat pessimistic. Do I believe that we can manage aging and extend the life of nuclear power plants beyond forty years? The answer is yes! But to do so it will take a complementary, coordinated effort on the part of researchers, utilities, NSSS vendors, architect engineers and regulators. The NRC's Nuclear Plant Aging Research Program along with other national efforts, industry efforts, and a general improvement in plant performance and maintenance can provide a solid basis for life extension and management of aging. Obviously, these must be done well. They must be comprehensive and not driven to premature conclusions by a license expiration date.

Actually, as you know even without life extension we need to aggressively pursue aging assessment and management. As you know, some aging begins when parts and components are made and it continues for the life of the plant. Such aging can, if not managed, impact safety through simultaneous failures of redundant systems or components and result in reduction in defense-in-depth. So we have no choice but to manage aging and manage it well. In the United States I believe we have made great strides in the last few years in improving the operation, testing, and maintenance of the plants.

Through the efforts of the utilities, industry organizations, vendors and the NRC, I see significant improvements in maintenance, testing, plant material condition, trending and failure analysis, operator performance, emergency and operating procedures and other areas. With continuing further improvement in these areas coupled with a good maintenance program to manage aging and with a sound national and international aging research program, I am optimistic that we can manage plants safely as they age and justify some life extension.

But in my view we have just begun!

NUCLEAR PLANT AGING: AN OWNER'S PERSPECTIVE

By Jack H. Ferguson,
President and Chief Executive Officer, Virginia Power

It is a pleasure to be here among so many distinguished representatives of the international nuclear industry.

My remarks today are from the perspective of an owner. My company, Virginia Power, has been an operator of nuclear plants for 16 years. We have four nuclear units, two each at our Surry and North Anna stations.

Last November, in testimony before the Subcommittee on Energy and Power of the U.S. House of Representatives Energy and Commerce Committee, I said, "In more than 46 reactor years of operation, we have experienced no inherent problem of aging or deterioration which would prevent any of our nuclear units from functioning safely and efficiently for the full 40 years for which they are licensed."

That statement remains true today.

I acknowledged that my company had experienced two major events at our nuclear stations within the year before I testified. They were a feedwater pipe failure #1 Surry station and a steam generator tube rupture at North Anna.

Contrary to certain widely publicized but uninformed opinions, neither of those events was caused by normal or true aging. Both were examples of deterioration associated with specific sets of circumstances. The Surry pipe failure was caused by erosion/corrosion, and the North Anna tube rupture resulted from vibration-related metal fatigue.

Thorough NRC investigation and analysis corroborated our findings. Through coordinated industry efforts, improved maintenance and surveillance procedures reflecting information on those two potential problems are now in place not only at Virginia Power, but elsewhere in the U.S. and other countries.

Not only is my company confident that our stations will remain safe and efficient for their full initial license period, we expect to be able to extend their lives beyond 40 years.

Our Surry Unit 1 serves as one of two pilot plants in a study of the technical and economic feasibility of life extension. The study is principally funded by the U.S. Department of Energy and the Electric Power Research Institute. Phase One of the study has been completed, and the EPRI report concludes that (quote) "there appear to be no major obstacles to operating Surry Unit 1 for considerably longer than the current license term of 40 years."

Clearly, many components of a nuclear plant will not last 40 years. They were never designed to do so. As components individually wear or deteriorate, they will be replaced. We, and the rest of the nuclear industry, maintain stringent preventive or predictive maintenance programs to detect wear or deterioration. These are augmented by surveillance and non-destructive testing.

The industry's maintenance programs are continually evolving and improving. What is good enough today may well not be good enough tomorrow, as we learn more about the mechanisms of age-related degradation. But the need to change and upgrade systems and procedures is not unique to nuclear power.

Even more important, individual utilities are no longer dependent only on their own resources to solve operating and maintenance problems. The U.S. nuclear industry is no longer an industry of builders, but of operators. The industry has more collective experience in operating plants than ever before, and is sharing that experience more broadly than ever.

Our industry associations have been restructured to reflect the industry's reorientation toward operations. The Institute for Nuclear Power Operations (INPO) continues to set standards of operating excellence for the industry. More recently the Nuclear Management and Resources Council (NUMARC) assumed the role of working with the NRC to seek industry-wide solutions to technical and regulatory issues, including operations and maintenance questions. Nuclear plant life extension and all it entails is a priority issue with NUMARC.

The U.S. nuclear industry also shares information with other nations. My own company, for example, has exchange agreements with utilities in Japan, France, Switzerland and Spain. This conference is an excellent example of international information sharing.

It is clear that nuclear plants are aging--everything ages. But it is also clear that "aging" is not a new issue. It is one that is being addressed throughout the industry.

I believe that the safety implications of the so-called "aging" of U.S. nuclear plants are more than offset by the maturing of nuclear operators. The knowledge and ability to deal with this issue, insofar as it is an issue, is well within the industry's capability.

Klaus Gast
Director, Office for Safety of Nuclear Installations

Bundesministerium fur Umwelt, Naturschutz und Reaktorsicherheit
Federal Republic of Germany

Let me start with an overview of the approach on aging that we take in the Federal Republic of Germany with a statement that may sound almost trivial.
A nuclear power plant may age in the sense of accumulating years of operation, but it must stay in a safe condition according to the regulatory requirements until the end of its life.

And to turn this around, with regard to safety, the lifetime of a nuclear power plant is only determined by its safety status. So any measures taken to provide, maintain, and enhance the nuclear safety status of a plant, and in particular measures to prevent, mitigate, repair, or compensate aging degradation of properties relevant to safety, generally also effectively allow for lifetime extension.

With regard to such measures, one may differentiate between two principally different kinds of aging, physical and nonphysical aging. By nonphysical aging, I mean that a plant designed and built to yesterday's standards may progressively differ from the developing state of the art unless it is upgraded or backfitted accordingly. Physical aging is actually the theme of this symposium. I, myself, would define it as the accumulation of changes of safety-relevant properties of the plant components caused by ambient and operational influences.

Now the safety policy in the Federal Republic of Germany aims at minimizing both kinds of nuclear power plant aging, the nonphysical and the physical. Of course, with regard to risk, measures taken against nonphysical aging can counteract or may even outbalance the effect of physical aging.

This policy includes the following elements. First, aging-resistant and, let me use the phrase age-tolerant, design of vital components with adequate safety margins to account for aging effects. For instance, the so-called concept of basic safety as laid down in the Risk Guidelines for PWRs, limitation of the fast neutron flux (>1 MeV) to the vessel wall to a fluence of 10^{19} nvt, etc.

Second, extensive and nearly continuous supervision and control of the operational status of the plant by the state authority and/or their independent experts, notwithstanding, of course, the utility's own activities and responsibilities.

Third, continuous evaluation and feedback of operating experience. Nuclear power plants in the Federal Republic are erected by turnkey contracts. There is a high degree of standardization with respect to materials, design principles, and layout. Analysis of operating experience is a powerful tool to identify degradation of materials, components, and reliability of systems.

Fourth, continuous surveillance of safety systems and of operational conditions, optimization of systems, and of plant operating procedures. For instance, minimizing the number of SCRAMs and optimizing system reliability by in-service inspection and preventive maintenance according to the experience gained.

Fifth, systematic upgrading of the safety of plants. For instance, independent additional emergency systems; improvement of the emergency energy supply system and of the ECCS by hot leg injection; improvements with regard to accessibility and effectiveness of inspection, maintenance, repair or replacement of components.

Sixth, research and development work, including material research (and I refer to the presentation that was given by Professor Kussmaul two days before) on full-scale components as a basis for predicting component behavior at worst case conditions, including end of life extrapolations; developing more sophisticated inspection, testing, and monitoring methods capable of detecting relevant changes in material behavior or reliability of components; and quantification of safety margins.

Last but not least, I would like to mention that it is foreseen for the future to have regular recurring safety evaluations of the individual plants at reasonable time intervals. This evaluation will include all important aspects of physical and nonphysical aging.

In concluding, I want to state my conviction that with appropriate actions taken to achieve and maintain a high level of safety, plant aging does not necessarily restrict the lifetime of nuclear power plants to a certain period, for instance, forty years.

Eduardo Gonzalez Gomez
Vicepresidente

Consejo de Seguridad Nuclear, Spain

Ladies and gentlemen, I first want to thank the NRC for inviting our organization to participate in this symposium. You can be assured that, even from our small country that has 7500 megawatts of nuclear capacity, we fully support the activities in this field, and we participate in many of them.

With respect to what has been discussed this week, I would like to make a remark. From my point of view, the aging of components from normal operation of plants does not have to be connected directly with life extension. I think that the first one is mostly a technical issue. For the second, my understanding is that with safety requirements being abated that there will arise the need for more stringent requirements, and this issue I think has to be addressed on its own. It may be something very trivial, but we all agree that aging is mostly a normal process during operation. And so the utilities running important investment installations have to take good care of them.

I think that five topics have to be concerns in this respect, and so we have stressed it to the Spanish utilities in this manner:

First of all, monitoring of operational components and service inspection. Second, preventive and protective maintenance and an analysis of the results. Third, incident analysis. Fourth, quality assurance in the operation. And fifth, exchange of information.

The role of the regulatory organization is to oversee and to make sure that the utilities are performing these activities in a correct manner. I think that even if the nuclear industry is now more than thirty years old that we will have to work hard to reach the necessary data bank that will allow the complete use of realistic assumptions.

But what I will do is just ask questions to the floor that I think have been raised already in the meeting, but maybe those are questions that we would discuss afterwards. For example, are the operating conditions producing the foreseen effects on materials and components? What is the real influence of maintenance on the degradation of the equipment? How can the regulatory organizations regulate maintenance? I know that the NRC is working on that, and I think that it will be very interesting to see how we can focus on this aspect.

Of course, from the regulatory standpoint, we always have to look further to the safety of the plants. And in that, how are we going to follow the evolution of safety with aging of components? Of course, if lengthening of the operating life is being studied, this will be a very important issue. And how will we use aging in our PSA and PRA evaluation of the plants?

Another aspect to take into consideration is the evolution of inspection techniques, which become more and more accurate and precise as to the underlying types of defects that have not been suspected before. On the other side of this same picture is the big series of tests that have shown the need for an evolution of nondestructive examination and ultrasonic examination standards. This may create problems in components that up to now have not shown acceptable indications.

The last point that I want to touch is the issue raised by the evolution of the technology in the electronic and processing systems, and from a regulatory point of view or from a safety point of view, when may a substitution of a component of this sort be called for for safety reasons.

It is not my intention to be critical or pessimistic, but in such a controversial industry as nuclear, we have to always be ahead of the problems that may arise.

I want to thank again the NRC for the opportunity that we all have had in this meeting, and I hope that through national and international efforts, we will be able to reach conclusions to the questions that I have asked and to many others. That could be a matter of several meetings.

Byron Lee, Jr.
President and Chief Executive Officer
Nuclear Management and Resources Council

It's been an exciting first year for the "new" NUMARC organization. I think it's been a very successful one as well. We've taken on a number of tough technical issues with generic impact, and have already helped the industry resolve several of them.

NUMARC's basic objective is to draw upon the nuclear industry's knowledge and experience to further enhance nuclear plant safety and reliability and to provide a unified channel for communications between the industry and the NRC. We have the full involvement -- and commitment -- of the industry's leaders. All 54 utilities are represented on the Board along with representatives from the major architect-engineering firms and the four NSSS vendors.

NUMARC helped bring closure to both the station blackout and the single-phase erosion-corrosion issues during this first year. We expect to do the same on two or three other generic issues by the end of 1988.

In recent months, NUMARC, with the technical support of EPRI, has coordinated the industry's response to a variety of allegations of fraudulent mechanical and electrical components. Our member utilities dedicated well over 100 man-years in the first six weeks to an aggressive and well-coordinated program of inspection, testing, and analysis. Our investigations so far have revealed no generic safety problems for the industry, but we recognize the potential exists for real problems and we are continuing to study this problem to determine appropriate long-term corrective action. We also have major efforts moving forward on issues that will be with us for many years, such as severe accidents, maintenance, and plant life extension.

NUMARC interest in aging and aging research relates to our efforts in the areas of maintenance and plant life extension. Our increasing understanding of the aging process enables us to detect age-related equipment problems and to replace equipment that can no longer perform its designed function. That's important to safety and to reliability.

Aging is important as we explore the requirements for extending our operating licenses beyond the initial 40-year license period. NUMARC's NUPLEX Working Group has just been formed and merged with the NUPLEX organization formed several years ago. The Working Group will coordinate the industry's efforts in life extension and expand the industry's involvement. We can't afford to wait until our plant licenses are about to expire before we tackle the many regulatory and technical issues, including aging, that are involved in extending those licenses.

Existing research on aging already has given us a large body of knowledge about its effects on the equipment that is replaced or refurbished as part of routine maintenance of our operating plants. The aging of these types of equipment is important but should not be a life extension concern. However, life extension raises new considerations of aging effects, particularly the aging of items such as pressure vessels, containments, and other components that are not expected to be replaced during a plant's initial license period.

There are several questions we need to answer to reap the large benefits available through extending plant life. What are the effects of long-term aging on these items? How can we mitigate those effects? What is the realistic life expectancy of this equipment, given appropriate maintenance?

As the industry continues its work in plant life extension, we will look to additional research on aging to help us answer these and other questions. We must be careful we don't give the impression to the public that we have reduced confidence in our present knowledge.

James Moore
Vice President
Westinghouse Electric Corporation

It is a pleasure for me to be here to have an opportunity to say a little bit to you on life extension from a supplier's viewpoint.

The first issue that I would like to address is one of timeliness. I think that it is very important that we move this life extension process forward expeditiously, and that we do not wait for the ultimate crisis to come as we are so wont to do on some of the other issues that we face as an industry.

In the year 2010, the majority of our operating plants in the United States will have reached the end of their licensed life. And we need a decision well before that forty-year mark to make the appropriate modifications or whatever for life extension and to determine alternatives or replacement decisions if they are needed. We just cannot wait until the last minute.

We also need to get a good data base, and we need that data base now for ultimate life extension decisions. If any of you have taken the time to try to resurrect and go back and find out information, whether it be design basis or transients that occurred five or ten years ago, I think that you will understand the need to really get a handle on a good data base, to manage it, to get it in a form that is easily accessible, and to get all the proper information.

We need to define the regulatory process, and we need to do that in a timely way right up front. And we need to define the technical priorities to solve the really key issues. To build on what Del Bunch said, we should not get hung up on overemphasis on the science of aging, but we should really deal with the key technical issues.

I have been a participant and an onlooker at the regulatory process for over thirty years now, and I think that today we have an opportunity to manage the life extension process in an orderly way. But we need good up-front NRC-industry cooperation to do that, and I am very pleased that the NRC has sponsored this symposium. The turn-out at the symposium I think is excellent, and it is a very good start. We need to build on this, and we need to coordinate our R&D programs internationally. We all have funding and budget problems, and we need to take advantage of each other collectively as an organization.

We have the mechanisms in place to manage this. We have EPRI, we have DOE, we have NUMARC, the NRC, the owners' groups. It is a mechanism that we have not had in the past, these various activities in such a form. We need to define responsibilities and set goals, and then manage these activities, and that is a responsibility for each of those organizations to set down what it is that they are going to do and then do it on schedule.

The lead plant approach that EPRI and DOE are working on, and part of which Jack Ferguson with Virginia Power talked about, is an excellent program, and I think that it is an excellent way to try to focus on and come to resolution of some of these issues by having lead plants actually go through the process. It is a very positive step.

International cooperation is absolutely mandatory. We have had good experience with our associates around the world. There is a huge data base that we all must share.

Life extension is a strategic issue. I would suggest that every utility, from the beginning, set up a dedicated resource that will define a life extension strategy for their plants. Then it needs to be phased into their normal operations. The purpose of laying out a long-term strategy is to maximize the cost effectiveness of the program and to integrate the life extension activities with the scheduled outages.

You may not think of it, but when a plant reaches mid-life, it may have as few as eleven or fifteen refueling outages left before it has reached the end of its licensed life. That is not a lot of time to accomplish the kinds of things you may need to do in the process of license extension, and doing it in a very cost effective way by doing it as part of plant outages.

You need to tie life extension into the operation and maintenance. It is an augmentation. There ought to be a way to actually save O&M costs. We are all worried about the O&M costs increasing considerably. I think that there is a way, by dealing with predictive maintenance versus preventive maintenance and doing it as a part of a total strategic life extension, that we can actually reduce operating and maintenance costs. And we clearly ought to improve availability. Life extension just necessarily has to help long-term availability.

What can a supplier's role be in this? I think that there are some obvious ones. We do have, of course, a large data base ourselves. We have design bases for these plants. We have a strong analytical capability and are continuing to expand that and improve our analytical techniques. We are very encouraged on the structural reliability and risk assessment approach that can be used to get a better feel for uncertainties and the impact on aging that was discussed yesterday in a paper that Ted Meyer gave.

Obviously, suppliers can help to keep plants running well, and that is key for all of the reasons that have been stated earlier. It is key in terms of not putting the plant through large transients, etc. We can be a big help in that.

In particular, we should be looking at the root cause for failures. I do not think that we as an industry in the U.S. do a very good job of, when we have a failure, really taking the time in a disciplined way to go down and actually determine the root cause of these failures. Too often, we do not go the whole ten yards.

We can find ways to avoid replacement of major components. We have a process that has been very effective to maintain steam generators on line without replacing them. We worked with our fuel cycle and fuel planning staff to reduce the fluence on the vessel from an NDT standpoint, and at the same time get improved fuel performance. These again are obvious ways that a supplier can help.

We as an industry can effectively reap the continued benefits of this very large installed base of nuclear plants. But it is really up to us collectively.

WORKING TOGETHER FOR SAFE, RELIABLE LONGEVITY OF NUCLEAR POWER

John J. Taylor
Vice President, Nuclear Power
The Electric Power Research Institute

Many life extension studies are underway by the utility industry, NSSS vendors, DOE and the NRC. We have heard about many of them at this symposium. These studies address material condition and aging of key systems, components, and structures and provide important information to assist utilities in effectively managing aging. No study has concluded that aging effects are a technical barrier to safe, economical operation of nuclear plants beyond their current licensed terms; i.e., the industry is monitoring aging effects and is in a position to manage them—it is now a matter of implementing initiatives that address effects that may become significant during extended life. The economic incentives for these initiatives are substantial.

EPRI is supporting the NUMARC NUPLEX Working Group to define these initiatives and has an extensive research program in cooperation with DOE which includes ongoing studies of aging effects.

In addition to research, operating experience augments our understanding of aging (radiation embrittlement, IGSCC, etc.). All in all, our understanding improves with age.

Keys to understanding and managing aging are:

PREVENTIVE AND PREDICTIVE MAINTENANCE PROGRAMS designed or improved for effective management of significant age-related degradation mechanisms,

VIGILANCE while using inspection and monitoring activities in anticipation of life extension, and

DATA TRENDING AND RESPONSIVENESS to findings as the experience database grows.

In a recent briefing to the Commission by Brian Sheron, it was pointed out that gaps in knowledge of aging effects could lead to regulatory conservatism and that an important ingredient is research by industry. I would like to challenge all interested parties—regulators, utilities, concerned public—to work together to identify significant gaps now so that EPRI and other industry groups can concentrate on them in future aging related research.

An excellent vehicle for identifying those gaps is the extensive NRC NPAR program. Utilities, first through the Equipment Qualification Advisory Group and now also through the NUMARC NUPLEX Working Group, have a dialog with the NRC and its national laboratory contractors through review and comment on the NPAR findings. It is imperative that the review and comment process be taken seriously by both the industy and NRC.

We should all search for excellence in the areas of:

accurately assessing the significance of aging effects that we don't fully understand,

continuing research in areas where greater understanding is required,

continuing to assess the results of utility aging management programs and enhance them where economically justified.

It is important that we distinguish research activities associated with aging from those overall activities which have as a goal the achievement of nuclear plant life extension. Life extension is directed primarily at those components and systems which are critical to safety and reliability and impractical or extremely expensive to replace. Preeminent examples of these are the containment and the reactor pressure vessel. But many components in the plant can be replaced economically and, in fact, will be replaced in the course of time either because of predicted aging degradation or economic obsolescence. It would therefore be a waste of precious R&D resources to probe into their extended lifetime when it is not required that they have such extended lifetime. An effective aging program will be one which focuses on the critical equipment and subsystems which should not be replaced if plant life extension is to be economically viable.

The payoff of all this effort and expenditure will be additional years of safe, reliable, and economical nuclear power generation.

AUTHOR AFFILIATIONS

J. S. Abel
Commonwealth Edison Company
One First National Plaza
P. O. Box 767
Chicago, IL 60690

Satish K. Aggarwal
General Chairman - International Nuclear
Power Plant Aging Symposium
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mamoru Akiyama
Professor
Department of Nuclear Engineering
University of Tokyo
7-3-1, Hongo, Bunkyo-ku
Tokyo 113, Japan

G. E. Apostolakis
School of Engineering and Applied Science
5532 Boelter Hall
University of California
Los Angeles, CA 90024

Guy A. Arlotto
Director
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Y. Asada
Professor
Department of Mechanical Engineering
University of Tokyo
Hongoh 7-3-1, Bunkyoku
Tokyo 113, Japan

Emil Bachofner
Manager
Nuclear Safety and Quality Assurance
OKG Aktiebolag
S-570 93 Figeholm
Sweden

M. L. Badlani
Manager
O'Donnell & Associates, Inc.
241 Curry Hollow Road
Pittsburgh, PA 15236

K. R. Balkey
Principal Engineer
Westinghouse Electric Corporation
P. O. Box 2728
Pittsburgh, PA 15230

B. Barthelet
Engineer
Service de la Production Thermique
Electricite de France
71 Rue de Miromesnil
75008 Paris
France

M. S. Bauge
Head of Bureau de Controle
de la Construction Nucleaire
Department of Industry
10 Boulevard Carnot
Immeuble Le Richelieu
21000 Dijon
France

Eric S. Beckjord
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Wallace B. Behnke, Jr.
Vice Chairman
Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Victor Benaroya
Assistant for Technical Studies
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Frank J. Berté
Supervisor, Statistical Engineering Services
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, CT 06095

Jerzy Bielanik
Technical University of Warsaw
Plock, ul. Lukasiewicza 17
Poland

Gerald Bimont
Research and Development Division
Electricite de France
3 rue de Mesine
75384 Paris Cedex 08
France

B. A. Bishop
Pricipal Engineer
Westinghouse Electric Corporation
P. O. Box 2728
Pittsburgh, PA 15230

C. B. Bond
Senior Engineer
Power Systems Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, PA 15230

Bruce M. Boyum
Principal Mechanical/Chemical Engineer
Mail Drop 981F
Washington Public Power Supply System
P. O. Box 968
3000 George Washington Way
Richland, WA 99352

R. J. Brandon
Manager, Engineering Analysis Services
General Electric - Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

A. M. Bruning
Lectromechanical Design Co.
Herndon, VA 20271

Delbert F. Bunch
Principal Deputy Assistant Secretary
for Nuclear Energy
U.S. Department of Energy
1000 Independence Avenue
Washington, DC 20585

J. J. Burns
Senior Mechanical Engineer
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Spencer H. Bush
Senior Consultant
Battelle-Pacific Northwest Laboratory
902 Battelle Boulevard
Richland, WA 99352

Stephen T. Byrne
Consulting Engineer
Combustion Engineering, Inc.
P. O. Box 500
Windsor, CT 06095

F. J. Campbell
Naval Research Laboratory
Washington, DC 20375

Nathan G. Cathey
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

Bindi Chexal
Manager, Erosion-Corrosion Program
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Madeleine Conte
Head of Section of Regulation, Documentation,
Foreign Relations
Safety Analysis Department
Institute of Protection and Nuclear Safety
Commissariat a l'Energie Atomique
B.P. No. 6
92265 Fontenay-aux-Roses Cedex
France

Gerald Cordier
Departement Materiel
Division Chaudieres Nucleaires
Electricite de France
3 rue de Mesine
75384 Paris Cedex 08
France

A. Bert Davis
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

G. Deletre
Engineer
Safety Analysis Department
Institute of Protection and Nuclear Safety
Commissariat a l'Energie Atomique
B.P. No. 6
92265 Fontenay-aux-Roses Cedex
France

Donald W. Edwards
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, MA 01701

Jack H. Ferguson
President and Chief Executive Officer
Virginia Power
P. O. Box 26666
Richmond, VA 23261

J. Fohl
Staatliche Materialpruefungsanstalt
Universitat Stuttgart
Pfaffenwaldring 32
7000 Stuttgart 80
Federal Republic of Germany

Joseph R. Fragola
Division Manager
Science Applications International Corp.
324 Madison Avenue
New York, NY 10173

Klaus Froehlich
Manager, Chemical Engineering
Bechtel - KWU Alliance
15740 Shady Grove Road
Gaithersburg, MD 20877

Klaus Gast
Director, Directorate Safety of
Nuclear Installations
Bundesministerium fur Umwelt
Naturschutz und Reaktorsicherheit
Postfach 12 06 29
5300 Bonn 1
Federal Republic of Germany

Bogdan Glumac
"Joseph Stefan" Institute
Reactor Physics Division
Ljubljana 61000
Jamova 39 (POB 100)
Yugoslavia

Eduardo Gonzalez Gomez
Vicepresidente
Consejo de Seguridad Nuclear
Madrid
Spain

G. M. Gordon
Manager, Materials Technology Products
and Services
General Electric - Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

W. R. Greenaway
Manager, Chemical Engineering
Bechtel - KWU Alliance
15740 Shady Grove Road
Gaithersburg, MD 20877

J. C. Guilleret
Plant Safety Attaché
Chooz A Power Station
Sena BP no. 60
08600 Givet
France

E. J. Hampton
Vice President
O'Donnell & Associates, Inc.
241 Curry Hollow Road
Pittsburgh, PA 15236

T. Hardin
Robert L. Cloud & Associates, Inc.
125 University Avenue
Berkeley, CA 94710

H. M. Hashemian
President
Analysis and Measurement Services Corporation
9111 Cross Park Drive, NW
Knoxville, TN 37923

K. Hattori
Heavy Apparatus Engineering Laboratory
Toshiba Corporation, 2-4 Suehiro-cho
Tsurumi-ku, Yokohama, 230
Japan

Takuya Hattori
Manager
Nuclear Power Plant Engineering Division
Tokyo Electric Power Co.
1-1-3 Uchisaiwai-cho, Chiyoda-ku
Tokyo 100
Japan

F. Hedin
Engineer
Service Etudes et Projets
Thermiques et Nucleaires
Electricite de France
12-14 Avenue Dutriévoz
69628 Villeurbanne Cedex
France

J. Y. Henri
Engineer
Safety Analysis Department
Institute of Protection and Nuclear Safety
Commissariat a l'Energie Atomique
B.P. No. 6
92265 Fontenay-aux-Roses Cedex
France

John M. Hicks
Senior Engineer
Westinghouse Semiconductor Control Center
Pittsburgh, PA 15238

J. P. Higgins
PLEX Program Manager
General Electric - Nuclear Energy
Mail Code 771
175 Curtner Avenue
San Jose, CA 95125

R. W. Hockenbury
Rensselaer Polytechnic Institute
JEC 5046
Troy, NY 12180

Kenneth R. Hoopingarner
Senior Research Scientist
Pacific Northwest Laboratory
P. O. Box 999
Richland, WA 99352

K. Iida
Professor
Department of Mechanical Engineering
Shibaura Institute of Technology
3-9-14 Shibaura, Minatoku
Tokyo 108
Japan

R. H. Jabs
Principal Engineer
Westinghouse Nuclear Services Division
P. O. Box 355
Pittsburgh, PA 15230

A. B. Johnson, Jr.
Pacific Northwest Laboratory
P. O. Box 999
Richland, WA 99352

D. P. Jones
Bettis Atomic Power Laboratory
Westinghouse Electric Corporation
West Mifflin, PA 15122

H. Kashiwaya
Heavy Apparatus Engineering Laboratory
Toshiba Corporation, 2-4 Suehiro-cho
Tsurumi-ku, Yokahama, 230
Japan

S. Kasturi
MOS
25 Piermont Drive
Melville, NY 11747

R. F. Kirchner
Rensselaer Polytechnic Institute
Troy, NY 12180

E. Kiss
Manager, BWR Technology
General Electric - Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

Y. Kitsunai
Senior Researcher
Mechanical Engineering Research Division
National Research Institute of Industrial Safety
1-4-6 Umezono
Kiyose 204
Japan

Leonard V. Konstantinov
Deputy Director General
Head, Department of Nuclear Energy and Safety
International Atomic Energy Agency
Wagramerstrasse 5, P.O.B. 100
A-1400 Vienna
Austria

Karl Kussmaul
Professor
Director, Staatliche Materialpruefungsanstalt
Universitat Stuttgart
Pfaffenwaldring 32
7000 Stuttgart 80
Federal Republic of Germany

Michel J. Lavérie
Director
Central Service for the Safety of
Nuclear Installations
Ministry of Industry
99, rue de Grenelle
75700 Paris
France

Byron Lee, Jr.
President/Chief Executive Officer
Nuclear Management and Resources Council
1776 Eye Street, NW
Washington, DC 20006

Daniel F. Lehnert
Vice President
Multiple Dynamics Corporation
29200 Southfield, Suite 103
Southfield, MI 48076

R. Lofaro
Research Engineer
Brookhaven National Laboratory
Upton, NY 11973

Philip E. MacDonald
Manager of Systems Safety and Risk Evaluation Group
Idaho National Engineering Laboratory
P. O. Box 1625
Idaho Falls, ID 83415

Shigero Masamori
Mitsubishi Heavy Industries, Ltd.
Kobe Shipyard & Machinery Works
1-1-1, Wadasaki-cho, Hyogo-ku
Kobe 652
Japan

H. W. Massie, Jr.
Manager, PLEX
Power Systems Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, PA 15230

Babette M. Meale
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

T. A. Meyer
Manager, Structural Materials and
Reliability Technology
Westinghouse Electric Corporation
P. O. Box 2728
Pittsburgh, PA 15230

Hiromichi Mitsuda
Manager, Maintenance Projects
Nuclear Power Operations Department
The Kansai Electric Power Co., Inc.
Nakanoshima, 3-chome
Kita-ku, Osaka
Japan

K. Miya
Professor
Nuclear Engineering Research Laboratory
Faculty of Engineering, University of Tokyo
2-22 Shirakata-shirane, Tohkai-mura, Nakagun
Ibaragi 319-11
Japan

K. Mokhtarian
Senior Engineer
Chicago Bridge & Iron Co.
800 Jorie Boulevard
Oak Brook, IL 60521

James S. Moore
President
Westinghouse Savannah River Co.
1359 Silver Bluff Road
Aiken, SC 29801

M. F. Moylan
Project Engineer
Wisconsin Electric Power Company
Nuclear Plant Engineering and Regulation Section
231 West Michigan Street
Milwaukee, WI 53201

Thomas E. Murley
Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mitja Najzer
"Joseph Stefan" Institute
Reactor Physics Division
Ljubljana 61000
Jamova 39 (POB 100)
Yugoslavia

Gerry Neils
General Manager
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

S. Novak
Division of Nuclear Safety
International Atomic Energy Agency
A-1400 Vienna, Austria

William J. O'Donnell
President
O'Donnell & Associates, Inc.
241 Curry Hollow Road
Pittsburgh, PA 15236

N. Ohtsuka
Engineering Consultant
Technology Department
Engineering and Operation Coordination Division
Chiyoda Corporation
2-12-1 Tsurumichuo, Tsurumiku
Yokohama 230
Japan

A. Okamoto
Manager, Research and Development Group
Engineering Administration and Development Department
Nuclear Power Division
IHI Corporation
1 Shinnakaharacho, Isogoku
Yokohama 235
Japan

Richard Orr
Westinghouse Electric Corporation
PSD, P. O. Box 2728
Pittsburgh, PA 15235

Dr. J. Pachner
Regulatory Research Branch
Atomic Energy Control Board
Ottawa, Canada K1P 5S9

W. Pavinich
Tennessee Valley Authority
400 W. Summit Hill Drive
Knoxville, TN 37902

K. M. Petersen
Operations Manager
Analysis and Measurement Services Corporation
9111 Cross Park Drive, NW
Knoxville, TN 37923

Terry Pickens
Senior Engineer
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

J. S. Porowski
Executive Vice President
O'Donnell & Associates, Inc.
241 Curry Hollow Road
Pittsburgh, PA 15236

Narendra Prasad
Westinghouse Electric Corporation
PSD, P. O. Box 2728
Pittsburgh, PA 15235

S. Ranganath
Manager, Materials Monitoring and
Structural Analysis
General Electric - Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

Philippe Revel
Manager
Nuclear Engineering Branch
Proposal Department
Framatome
92084 Paris la Defense
France

Jerral E. Rhoads
Manager, Equipment Engineering
Mail Drop 981F
Washington Public Power Supply System
P. O. Box 968
3000 George Washington Way
Richland, WA 99352

Kenneth C. Rogers
Commissioner
U.S. Nuclear Regulatory Commission
Washington, DC 20555

J. K. Rothert
Northeast Utilities Service Co.
P. O. Box 270
Hartford, CT 06101

Bernard C. Rudell
Supervisor, Materials Engineering - Nuclear
Baltimore Gas & Electric Co.
Lusby, MD 20657

D. L. Sanzo
School of Engineering and Applied Science
5532 Boelter Hall
University of California
Los Angeles, CA 90024

Takeshi Satoh
Mitsubishi Heavy Industries, Ltd.
Kobe Shipyard & Machinery Works
1-1-1, Wadasaki-cho, Hyogo-ku
Kobe 652
Japan

David G. Satterwhite
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

R. J. Schmidt
Northeast Utilities Service Co.
P. O. Box 270
Hartford, CT 06101

William L. Server
Manager of Material Engineering
Robert L. Cloud Associates
125 University Avenue
Berkeley, CA 94710

Vikram N. Shah
Principal Engineer
Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

Ulf Sjöö
Manager
Safety Review
OKG Aktiebolag
S-570 93 Figeholm
Sweden

George E. Sliter
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

P. P. Stancavage
Project Manager, BWR Life Extension
General Electric - Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

M. Subudhi
Technical Advisor
Brookhaven National Laboratory
Upton, NY 11973

Eugeniusz Szpunar
Institute of Atomic Energy
PL-05-400
Swierk-Otwock
Poland

K. Tajima
Heavy Apparatus Engineering Laboratory
Toshiba Corporation, 2-4 Suehiro-cho
Tsurumi-ku, Yokohama, 230
Japan

John J. Taylor
Vice President, Nuclear Power
The Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Jim Thomas
Duke Power Company
422 South Church Street
Charlotte, NC 28242

Brian Tomkins
U.K. Atomic Energy Authority
Risley, Warrington
Chesire WA3 6AT
United Kingdom

M. Tsubota
Heavy Apparatus Engineering Laboratory
Toshiba Corporation, 2-4 Suehiro-cho
Tsurumi-ku, Yokohama, 230
Japan

Hideo Uchida
Chairman
Nuclear Safety Commission
2-2-1 Kasumigaseki, Chiyoda-ku
Tokyo 100
Japan

N. Urabe
Senior Research Engineer
Steel Research Center
NKK Corporation
1-1 Minamiwataridacho, Kawasakiku
Kawasaki 210
Japan

Milton Vagins
Branch Chief
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

William E. Vesely
Senior Staff Scientist
Science Applications International Corp.
2941 Kenny Road, Suite 210
Columbus, OH 43221

J. P. Vora
Senior Electrical Engineer
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Gerald H. Weidenhamer
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

John Wreathall
Office Director
Science Applications International Corp.
2929 Kenny Road
Columbus, OH 43221

Chiaki Yasuda
Mitsubishi Heavy Industries, Ltd.
Takasago R&D Center
2-1-1, Shinhama, Arai-cho
Takasago 676
Japan

☆U.S.GOVERNMENT PRINTING OFFICE:1989-242-863/08157

NRC FORM 335
(8-87)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER *(Assigned by PPMB: DPS, add Vol. No., if any)*
NUREG/CP-0100

2. TITLE AND SUBTITLE
Proceedings of the International Nuclear
Power Plant Aging Symposium

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
February	1989

5. AUTHOR(S)
Ann F. Beranek, Compiler and Editor

6. DATE REPORT ISSUED

MONTH	YEAR
March	1989

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Conference Proceedings

b. PERIOD COVERED *(Inclusive dates)*
August 30-31,
September 1, 1988

12. SUPPLEMENTARY NOTES

13. ABSTRACT *(200 words or less)*

This report presents the proceedings of the International Nuclear Power Plant Aging Symposium that was held at the Hyatt Regency Hotel in Bethesda, Maryland, on August 30-31 and September 1, 1988. The Symposium was presented in cooperation with the American Nuclear Society, the American Society of Civil Engineers, the American Society of Mechanical Engineers, and the Institute of Electrical and Electronics Engineers. There were approximately 550 participants from 16 countries at the Symposium.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
plant aging

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

multiple components: 9-15 (japan)
27-38
183-197
201-206
241-249

Concrete : 84-88
17-14-2 PH SS: 77-83
Cast martensitic SS: 89-94
Erosion/Corrosion : 95-99
fatigue + crack growth : 100-113
elastomers : 118-124

Electrical components:
wiring : 130-136
microprocessors + IC's : 146-152
diesel generators : 153-157
Cable : 158-165

A533B PV steel : 207-211
Coolant pumps: 212-219
pressure vessel embrittle: 338-341
steam generators : 342-346
BWR internals : 347-350
CASS : 353-362
Resist. temp. detectors: 363-366

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC

PERMIT No. G-67