

NUREG-1144  
Rev. 2

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# Nuclear Plant Aging Research (NPAR) Program Plan

Status and Accomplishments

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**U.S. Nuclear Regulatory Commission**

Office of Nuclear Regulatory Research



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# Nuclear Plant Aging Research (NPAR) Program Plan

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Division of Engineering  
Office of Nuclear Regulatory Research  
U.S. Nuclear Regulatory Commission  
Washington, DC 20555



## ABSTRACT

A comprehensive Nuclear Plant Aging Research (NPAR) Program was implemented by the U.S. NRC Office of Nuclear Regulatory Research in 1985 to identify and resolve technical safety issues related to the aging of systems, structures, and components in operating nuclear power plants. This is Revision 2 to the Nuclear Plant Aging Research Program Plan. This plan defines the goals of the program, the current status of research, and summarizes utilization of the research results in the regulatory process. The plan also describes major milestones and schedules for coordinating research within the agency and with organizations and institutions outside the agency, both domestic and foreign.

Currently, the NPAR Program comprises seven major areas: 1) hardware-oriented engineering research involving components and structures; 2) system-oriented aging interaction studies; 3) development of technical bases for license renewal rulemaking; 4) determining risk significance of aging phenomena; 5) development of technical bases for resolving generic safety issues; 6) recommendations for field inspection and maintenance addressing aging concerns; 7) and residual lifetime evaluations of major LWR components and structures. The NPAR technical database comprises approximately 100 NUREG/CR reports by June 1991, plus numerous published papers and proceedings that offer regulators and industry important insights to aging characteristics and aging management of safety-related equipment. Regulatory applications include revisions to and development of regulatory guides and technical specifications; support to resolve generic safety issues; development of codes and standards; evaluation of diagnostic techniques, (e.g., for cables and valves); and technical support for development of the license renewal rule.

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## FOREWORD

The U.S. Nuclear Regulatory Commission's (NRC's) hardware-oriented engineering research program for plant aging and degradation monitoring of components and systems was first discussed in the initial version of the program plan, NUREG-1144, issued in July 1985. The program is identified as the Nuclear Plant Aging Research (NPAR) Program. It was stated in the plan that NUREG-1144 would be a living document to be revised periodically. The revisions would reflect the experience gained from implementing the plan. To be incorporated were 1) comments received from within the NRC, industrial codes and standards committees, and from domestic and foreign organizations and institutions; 2) research results and experience gained from the utilization of the plan in the regulatory process. The first revision to the plan, NUREG-1144, Rev. 1, was issued in September 1987.

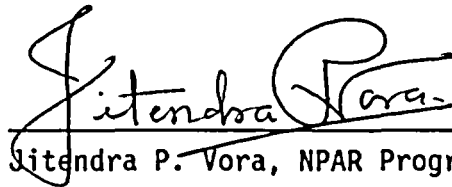
The Office of Nuclear Regulatory Research (RES) staff, which oversees the development and implementation of NUREG-1144, has since received numerous additional comments and updates on this document from various offices within the NRC and from individuals, organizations, and institutions outside NRC, both domestic and foreign. This revision reflects those comments.

The NRC provided planning guidance for needed safety research on plant aging and license renewal in its 1987 Policy and Planning Guidance document (NUREG-0885, Issue 5). The Executive Director for Operations (EDO) provided specific program guidance to the staff for FY 1986 to 1988 planning and program development. The NRC staff provided their comments on the current research program and needs for additional research and prioritization by "user-need" letters to RES and through the Technical Integration Review Group for Aging and Life Extension (TIRGALEX) review of the NPAR Program. More recent review and guidance have come from the NPAR Review Group, represented by staff from the NRC offices and from the Regions; from the Nuclear Safety Research Review Committee (NSRRC); and from the Advisory Committee on Reactor Safeguards (ACRS).

This document presents a revised research plan that addresses the identification and resolution of technical safety issues relevant to plant aging and license renewal. This document also describes the utilization of results of aging assessments in the regulatory process and presents the current status of major program activities and significant accomplishments. Additionally, this plan describes methods to mitigate the effects of age-related degradation on electrical and mechanical components and nuclear plant safety systems. These methods include inspection, diagnostics, condition monitoring, maintenance, trending, and recordkeeping.

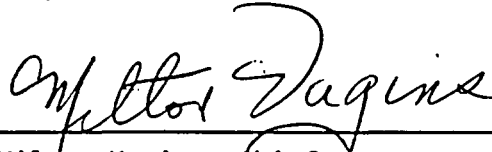
In conjunction with related program plans for primary system pressure boundary components and civil structures, this plan forms the overall framework for NPAR within the Division of Engineering, Office of Nuclear Regulatory Research of the NRC.

We welcome comments on this plan and will consider them in developing subsequent editions of this document. Comments need not be restricted to the research activities described herein; comments identifying omissions and/or recommending additional research are also welcome.



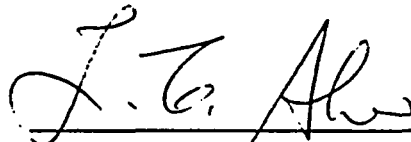
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## SUMMARY

The charter of the Nuclear Plant Aging Research (NPAR) Program described in this plan is to identify and address technical safety issues related to the aging of electrical and mechanical components and safety systems and support systems in commercial nuclear power plants (NPPs). The aging term of interest includes the period of normal licensed plant operation as well as the period of extended plant operation that may be requested in utility applications for license renewals. The principal goals of the program are to understand the effects of age-related degradation in NPPs and how to manage and mitigate them effectively.

The NPAR Program comprises seven major areas: 1) hardware-oriented engineering research involving components and structures; 2) system-oriented aging interaction studies; 3) development of technical bases for license renewal rulemaking; 4) risk significance of aging phenomena; 5) development of technical bases to resolve generic safety issues; 6) recommendations for field inspection and maintenance addressing aging concerns; 7) and residual lifetime evaluations of major light-water reactor (LWR) components and structures.

Significant progress has been made in completing the aging assessments of key components and systems since the prior revision of NUREG-1144. Nineteen safety-related components are under NPAR investigation, including five for which Phase-II assessments are completed. Seventeen safety-related systems are under NPAR investigation, including two for which Phase-II assessments are completed. Progress has also been made in developing models and approaches to evaluate the effects of age-related degradation on plant risk and effectiveness of maintenance to alleviate aging concerns. Studies to identify degradation sites and life-limiting processes for each major component have produced useful data. The Shippingport Aging Evaluation provided over 200 naturally-aged components and specimens of materials for aging evaluations. Operating experience and expert opinion have been applied to these aging assessments.

Initiatives to develop the technical bases for license renewal of NPPs have advanced since the last program plan update. On the regulatory side, plant license renewal has been identified as an important current NRC initiative. The NRC is developing a License Renewal Rule, and the NPAR Program has several initiatives underway to support the Rule, including development of a regulatory guide on Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses (DG-1009); technical support to the Office of Nuclear Reactor Regulations (NRR) in the following activities: development of Safety Evaluation Reports (SERs); development of a Standard Review Plan for License Renewal (SRP-LR); and reviews of lead plant license renewal applications; also, participation in reviews of Industry Reports. On the industry side, two lead plants are developing license renewal requests, planned for submission to NRC in 1991. The two plants are the Monticello boiling-water reactor (BWR) and the Yankee Rowe pressurized-water reactor (PWR). The license renewal initiatives are

sponsored by the U.S. Department of Energy, Electric Power Research Institute, and utilities, coordinated by the Nuclear Utility Management and Resources Council (NUMARC).

The NPAR staff has been prominent in international initiatives on NPP aging. A 1987 International Atomic Energy Agency (IAEA) symposium on NPP Aging and Life Extension in Vienna, Austria, was attended by delegates from 30 countries. The program included 14 presentations by NPAR speakers from a total of 40 papers. A major IAEA initiative, now underway on NPP aging, has been guided to a major extent by methods and results developed under the NPAR program. In August 1988, the NPAR Program hosted an international symposium on NPP aging in Bethesda, Maryland. Speakers from 10 nations participated, and a proceedings has been published (NUREG/CP-0100).

Through its research, the NPAR Program is providing information for timely and sound regulatory decisions on the operation of NPPs of all ages, including the possibility of extended operation. The research results are providing specific inputs to the regulations, including revisions to regulatory guides, technical specifications, generic safety issues, and codes and standards.

Future NPAR plans include the following:

- continuation of research on safety-related systems, structures, and components to address technical safety issues related to aging
- continuing studies on special topics, including aging effects on plant risk, and residual life assessments of major LWR components
- providing guidance for effective field inspections and maintenance that address understanding and managing aging in safety-related components and structures
- providing input to the license renewal process, including support to the License Renewal Rule, assistance with reviews of industry positions, and further resolution of aging issues that are specific to extended operation; support to NRR in developing SERs and the SRP-LR for license renewal
- application of NPAR strategy and database to assessment of aging impacts on mothballed plants
- support to develop the Maintenance Rule and a Regulatory Guide
- motivation to implement well-structured aging management programs
- application of lessons learned from NPAR aging assessments to advanced reactor technology.

## ACKNOWLEDGMENTS

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John Nageley and Robert Pedersen at PNL have provided editorial assistance.

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(a) Represented by the corresponding NPAR project managers.



## ACRONYMS AND INITIALISMS

ACRS	Advisory Committee on Reactor Safeguards
AEOD	Office of Analysis and Evaluation of Operational Data of NRC
AFW	auxiliary feedwater
ALEXCC	Aging and Life Extension Coordinating Committee
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American Nuclear Standards Institute
ASME	American Society for Mechanical Engineers
BNL	Brookhaven National Laboratory
BOP	balance of plant
BWR	boiling-water reactor
DOE	U.S. Department of Energy
ECCAD	electrical circuit characterization and diagnostic
ECCS	emergency core cooling system
EDG	emergency diesel generator
EDO	Executive Director for Operations
EIS	environmental impact statement
EMEB	Electrical and Mechanical Engineering Branch of the Division of Engineering
EPRI	Electric Power Research Institute
EQAG	Equipment Qualification Advisory Group of EPRI
GSI	generic safety issues
HPCI	high pressure coolant injection system (PWRs)
IAEA	International Atomic Energy Agency

I&C            instrumentation and control  
IEEE           Institute of Electrical and Electronic Engineers  
INEL           Idaho National Engineering Laboratory  
IPRDS          In-Plant Reliability Data System  
ISMM          inspection, surveillance, and monitoring methods  
IR             industry report  
LER           licensee event report  
LR             license renewal  
LWR           light-water reactor  
MCC           motor control center  
MEB           Materials Engineering Branch of the Division of Engineering  
MOV           motor operated valve  
NDE           nondestructive examination  
NIST           National Institute of Standards and Technology  
NOAC           Nuclear Operations Analysis Center at ORNL  
NPAR           Nuclear Plant Aging Research  
NPP           nuclear power plant  
NPRDS          Nuclear Plant Reliability Data System  
NRC           U.S. Nuclear Regulatory Commission  
NRR           Office of Nuclear Reactor Regulation of NRC  
NSAC           Nuclear Safety Analysis Center, operated by EPRI  
NSSS          nuclear steam supply system  
NUMARC       Nuclear Utility Management and Resources Council  
NUPLEX       Nuclear Utility Plant Life Extension working group of NUMARC  
ORNL           Oak Ridge National Laboratory

**PLEX**            **plant life extension**  
**PNL**             **Pacific Northwest Laboratory**  
**PORV**           **power-operated relief valve**  
**PRA**            **probabilistic risk assessment**  
**PSA**            **probabilistic safety analysis**  
**PWR**            **pressurized-water reactor**  
**RCIC**           **reactor core isolation cooling**  
**RES**            **Office of Nuclear Regulatory Research of NRC**  
**RHR**            **residual heat removal**  
**RPS**            **reactor protection system**  
**SSCs**           **systems, structures, and components**  
**SNL**            **Sandia National Laboratory**  
**SNUG**           **snubber utility group**  
**SRP-LR**        **Standard Review Plan for License Renewal**  
**SSEB**           **Structural and Seismic Engineering Branch of the Division of Engineering**  
**SWS**            **service water system**  
**SWWG**         **Service Water Working Group of EPRI**  
**TIRGALEX**     **Technical Integration Review Group for Aging and Life Extension**

## WORKING DEFINITIONS FOR NPAR<sup>(a)</sup>

- accelerated aging\* - synonym for artificial aging (when the simulation of natural aging is performed by application of stressors beyond those found in service in order to obtain observable aging effects in a reasonable period of test time).
- aging\* - showing the effects of time or use in the physical characteristics of a system, structure, or component.
- aging degradation\* - gradual deterioration in the physical characteristics of a system, structure, or component that is due to aging mechanisms, occurs with time or use under pre-service or service conditions, and could impair its ability to perform any of its design functions.
- aging management\* - engineering, operations, and maintenance activities to control aging degradation and failures due to aging of systems, structures, or components to within acceptable limits.
- aging mechanism\* - process that gradually changes the physical characteristics of a system, structure, or component with time or use.
- common cause failure\* - two or more redundant component failures due to a single cause (IEEE Standard 100).
- common mode failure\* - two or more redundant component failures in the same manner or mode.
- condition monitoring\* - continuous or periodic measurement and trending of the performance or physical characteristics of a system, structure, or component to indicate its current or future performance.
- extended operation - operation of a nuclear power plant beyond the term authorized by its current operating license.
- important to safety - defined by 10 CFR 50, Appendix A, as "Those structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." For details, see 10 CFR 50.49.

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(a) Starred entries were developed by a Committee on Common Aging Terminology with representatives from NRC, research laboratories, electric utilities, EPRI, and NUMARC. The initiative was sponsored by EPRI and was coordinated by MPR Associates. A report is expected to be issued in 1991.

- inspection\* - observation or measurement to verify that the physical characteristics of a system, structure, or component conform to acceptance criteria.
- license extension - authorized extension of the operating license to allow for the time elapsed between the dates of the construction permit (CP) and operating license (OL).
- license renewal - the issuance of an operating license which supersedes an existing operating license for a NPP.
- maintenance, corrective\* - actions that restore, by repair, overhaul, or replacement, the capability of a failed system, structure, or component to perform its design function within acceptance criteria.
- maintenance, predictive\* - a form of preventive maintenance performed periodically or continuously to monitor, inspect, test, diagnose, or trend a system's, structure's, or component's performance or condition indicators; results indicate or forecast functional ability or the nature and schedule of planned maintenance prior to failure.
- maintenance, preventive\* - periodic, predictive, or planned maintenance performed prior to failure of a system, structure, or component to extend its service life by controlling degradation or failure.
- natural aging\* - aging of a system, structure, or component which occurs under preservice and service conditions.
- plant life assessment - the process of ensuring that NPP components, systems, and structures can be maintained in a condition to safely and reliably perform their design functions throughout extended operation.
- safety-related items - defined by 10 CFR 100, Appendix A, as a subset of components "important to safety" (see definition above).

This definition includes those systems, structures, or components designed to remain functional in the event that a Safe Shutdown Earthquake occurs. The required safety functions include maintaining

1. the integrity of the reactor coolant pressure boundary
2. the capability to shut down the reactor and maintain it in a safe shutdown condition

3. the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guideline exposures of this part (10 CFR 100).

Note: Regulatory Guide 1.29, "Seismic Design Classification," lists light-water reactor SSCs that are to be designed to Seismic Category I and, therefore, are considered safety-related according to 10 CFR 100, Appendix A.

Note: The ASME Boiler and Pressure Vessel Code, in Section III, (ASME 1983), requires the classification of components into 3 classes. Class 1 is safety-related pressure retaining equipment; Class 2 is the pressure retaining equipment that is essential for safe operation of the plant, but does not, under normal operation contain nuclear materials; and Class 3 is for other equipment.

- surveillance\* - observation or measurement of the performance or physical characteristics of a system, structure, or component to verify that it conforms to acceptance criteria.
- trigger event - an operational transient or minor accident that can lead to a more serious event when followed by failures in safety-related systems. Emphasis is placed on the relationship between failures (causes and modes) expected to be experienced during operation and those that would potentially occur under the stresses associated with design basis or trigger events.

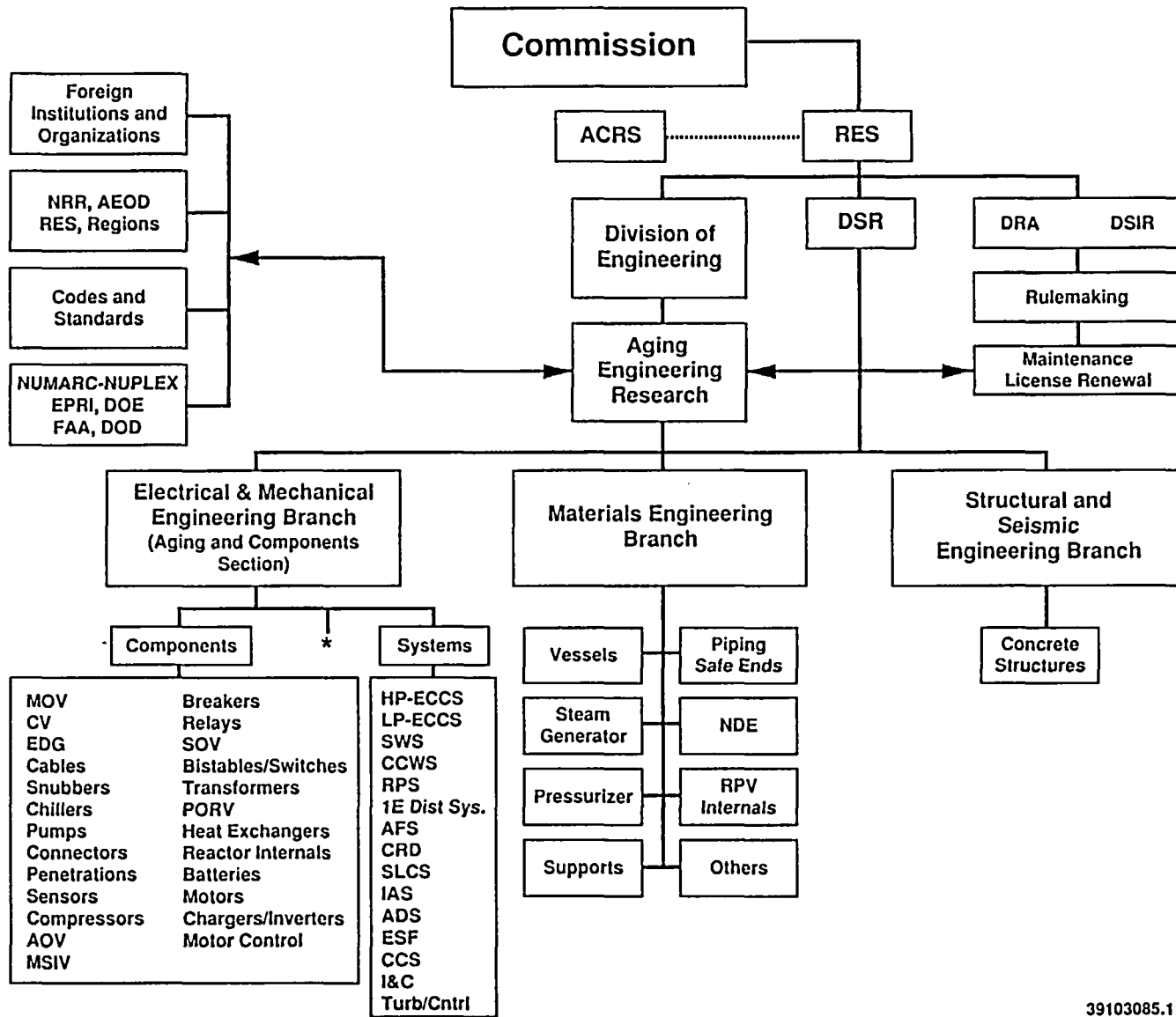
## 1.0 INTRODUCTION

The United States has more than 110 nuclear reactors in commercial operation and several of these reactors have operated more than 20 years. As the population of light-water reactors (LWRs) advances in age, the need for a research program that systematically assesses the effects of age-related degradation on the safety of these plants has been recognized. The Director of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), in his comments on the Long-Range Research Plan, identified a need for a research program to investigate the safety aspects of aging processes in commercial nuclear power plants (NPPs). The Advisory Committee on Reactor Safeguards (ACRS), in their 1983 report to Congress, also recommended initiating an aging research program.

The NRC provided guidance for needed safety research on plant aging and license renewal in its Policy and Planning Guidance document (NUREG-0885). Also, the Executive Director for Operations (EDO) provided specific program guidance to the staff for FY 1986 to 1988 planning and program development.

In response to this need, the NRC Office of Nuclear Regulatory Research (RES) developed and implemented a hardware-oriented engineering research program for understanding and managing aging of safety-related components and systems. This program is identified as the Nuclear Plant Aging Research (NPAR) Program. As shown in Figures 1.1 and 1.2, the program is conducted by the Electrical and Mechanical Engineering Branch of the Division of Engineering of the Office of Nuclear Regulatory Research. Similar programs are conducted on other aspects of aging in NPPs (Vagins and Taboada 1985; Muscara and Serpan 1985; Vagins and Strosnider 1985; Muscara 1985). One of these programs focuses on aging in NPP vessels, piping, and steam generators and non-destructive examination techniques, and is being conducted by the Materials Engineering Branch of the Division of Engineering (MEB). The program plan developed by MEB is a companion to the NPAR program plan. Another related program involves an assessment of the impact of age-related degradation on plant civil structures and is being conducted by the Structural and Seismic Engineering Branch (Naus et al. 1989). These three programs form the foundation for research on aging and are implemented within the Division of Engineering, Office of Nuclear Regulatory Research (NRR) of the NRC.

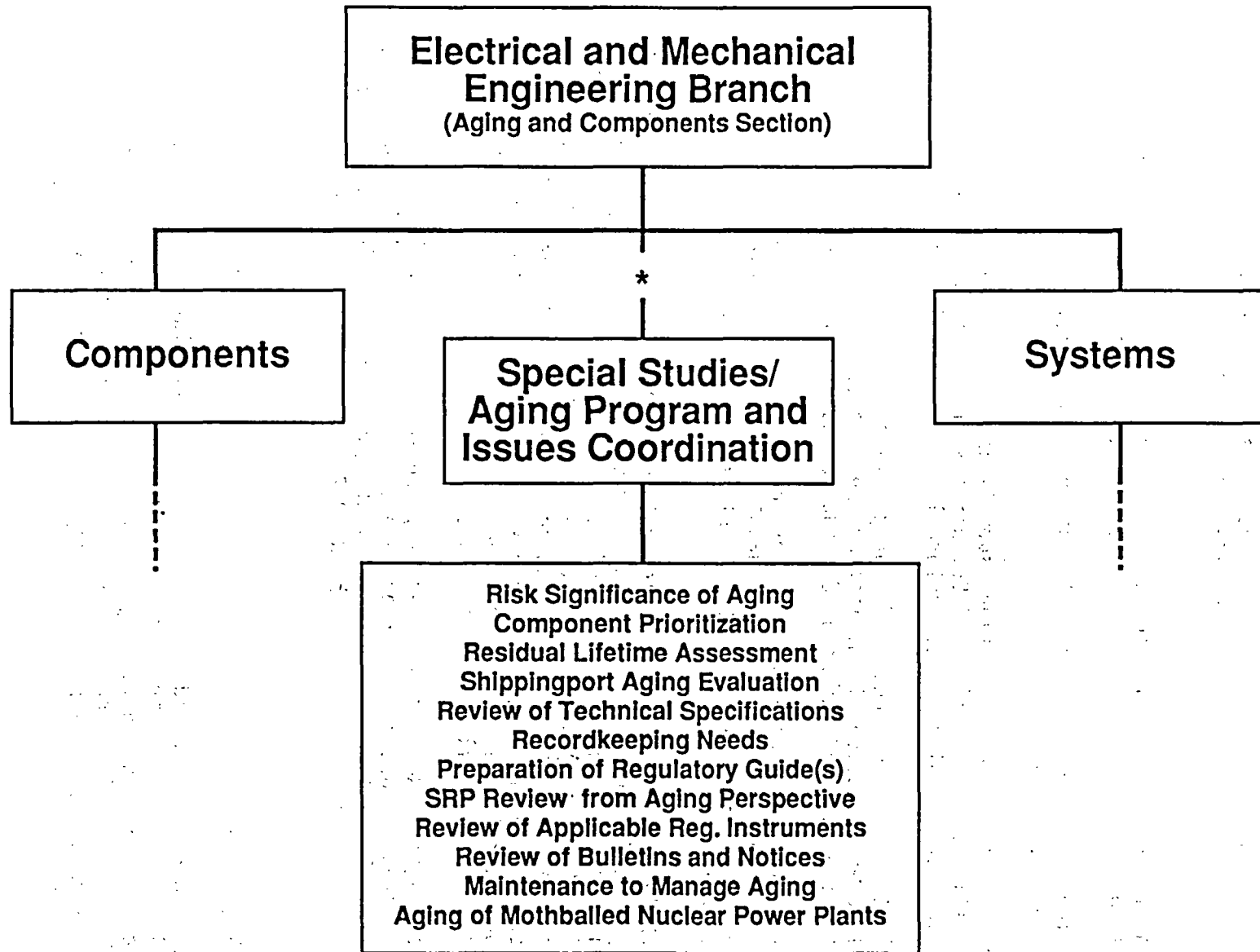
The NPAR Program Plan was first described in the July 1985 issue of NUREG-1144 (Morris and Vora 1985). In September 1987, the Program Plan was revised and reissued as NUREG-1144, Rev. 1 (Vora 1987). This report, Revision 2, describes the current NPAR Program Plan, discusses progress, status, and accomplishments in the program since issuance of Revision 1, and includes future plans. Revision 2 also emphasizes the research currently being conducted to resolve the technical safety issues relevant to the aging of NPPs, both in the current licensing period and in extended operation. The utilization of the NPAR technical data in the regulatory process is also included.



39103085.1

**FIGURE 1.1.** Aging Research Structure





1.3

39103085.2

**FIGURE 1.2.** NPAR Program Elements

## 1.1 GOALS AND OBJECTIVES OF THE NPAR PROGRAM

The main goals of the NPAR Program are to understand aging and to identify ways to manage aging of safety-related systems, structures, and components (SSCs) in NPPs. The following are the technical objectives of the Program:

- identify and characterize aging effects which, if unmitigated, could cause degradation of SSCs and thereby impair plant safety
- develop supporting data and information to facilitate management of age-related degradation
- identify methods of inspection, surveillance, and monitoring, or of evaluating residual-life of SSCs, which 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 effects of aging and diminishing the rate and extent of degradation caused by aging
- provide technical bases and support for the License Renewal Rule and the license renewal process and develop a regulatory guide on the format and technical information content for license renewal applications.

The NPAR Program has been developed to meet these goals and objectives. Currently, the NPAR Program comprises seven major areas: 1) hardware-oriented engineering research on components and structures; 2) systems-oriented aging interaction studies; 3) development of technical bases for license renewal rulemaking; 4) risk significance of aging phenomena; 5) development of technical bases for resolving generic safety issues; 6) recommendations for field inspection and maintenance to address aging concerns; and 7) residual lifetime evaluations of major LWR components and structures.

Other ongoing NRC programs, industry-sponsored research, and programs being conducted in foreign countries have been considered in developing the NPAR Program Plan. In cases where relevant results from aging studies are available or are being developed, the NPAR Program has been planned to avoid duplication of effort.

## 1.2 BACKGROUND AND NEED

The NPAR Program has identified aging as the cumulative, time-dependent degradation of a system, structure, or component in a NPP that, if unmitigated, could compromise continuing safe operation of the plant. Necessary measures must, therefore, be taken to ensure that aging does not reduce the operational readiness of a plant's safety systems and does not result in common-mode failures of redundant, safety-related equipment, thus reducing

defense-in-depth. It is also necessary to ensure that aging does not lead to equipment failure which could cause an accident.

In December 1990, there were 112 licensed commercial LWR power plants in operation in the U.S. The following summarizes the age distribution of these plants:

<u>Operating Lifetime (Years Since Operating License)</u>	<u>Number of Plants</u>
More than 20	12
Between 15 and 20	37
Between 10 and 15	16
Between 5 and 10	26
Less than 5	21

The 30 oldest operating plants are listed in Table 1.1; the two oldest, Yankee Rowe and Big Rock Point, have been in operation for 30 and 28 years, respectively. However, they are demonstration plants with design power of <200 MWe. The next oldest plant, San Onofre-1, has a net capacity of 436 MWe and has been licensed for 24 years. In addition to the plants in operation, there are 6 more plants under construction. Most of these plants are expected to be in operation within the next decade.

As the population of U.S. LWRs has aged, problems have occurred as a result of time-dependent degradation mechanisms such as stress corrosion, thermal aging, radiation embrittlement, fatigue, and erosion. These problems have included failures in pumps, valves, and relays, embrittlement of cable insulation, and cracking of the heat-treated anchor heads for post-tensioning systems in containment. Although progress is being made to mitigate the degradation that has already been identified, significant questions concerning age-related degradation of SSCs remain because of the variety of components in a commercial power reactor, the complexity of the aging process, and the limited experience with prolonged operation of these power plants.

The NPAR Program also provides key information to enable the NRC to resolve technical safety issues and define its policy and regulatory position on license renewal (LR). The License Renewal Rule (55 FR 29043) is being developed by the NRC to provide a clearly defined policy and regulatory position regarding the safe operation of aged plants for extended operation. Clearly defined policies and criteria are needed to ensure that requests for LR address the primary regulatory concerns and relevant technical safety issues. Reactors are licensed for up to 40 years of operation under current regulations. Current regulations also permit renewal of an operating license. The NRC Technical Integration Review Group for Aging and Life Extension (TIRGALEX), established in 1986, developed a working definition for LR.

TABLE 1.1. Schedule of License Expirations for the 30 Oldest Operating Plants

<u>No.</u>	<u>Name of Plant</u>	<u>PWR/BWR</u>	<u>Design Rating (Net MWe)</u>	<u>Date Operating License Issued</u>	<u>License Expiration</u>
1	Yankee Rowe	PWR	175	07/60	2000
2	Big Rock Point	BWR	72	08/62	2002
3	San Onofre-1	PWR	436	03/67	2007
4	Haddam Neck	PWR	582	06/67	2007
5	Nine Mile Point-1	BWR	620	08/68	2008
6	Oyster Creek-1	BWR	650	08/69	2009
7	Ginna	PWR	470	09/69	2009
8	Dresden-2	BWR	794	12/69	2009
9	Monticello-1	BWR	545	09/70	2010
10	Robinson-2	PWR	700	09/70	2010
11	Millstone-1	BWR	660	10/70	2010
12	Point Beach-1	PWR	497	10/70	2010
13	Dresden-3	BWR	794	03/71	2011
14	Surry-1	PWR	788	05/72	2012
15	Point Beach-2	PWR	497	05/72	2012
16	Turkey Point-3	PWR	693	07/72	2012
17	Pilgrim-1	BWR	655	09/72	2012
18	Palisades	PWR	805	10/72	2012
19	Quad Cities-2	BWR	789	12/72	2012
20	Quad Cities-1	BWR	789	12/72	2012
21	Surry-2	PWR	788	01/73	2013
22	Vermont Yankee	BWR	540	02/73	2013
23	Oconee-1	PWR	887	02/73	2013
24	Turkey Point-4	PWR	693	04/73	2013
25	Maine Yankee	PWR	825	06/73	2013
26	Fort Calhoun-1	PWR	478	08/73	2013
27	Prairie-1	PWR	530	08/73	2013
28	Indian Point-2	PWR	873	09/73	2013
29	Oconee-2	PWR	887	10/73	2013
30	Zion-1	PWR	1040	10/73	2013

License renewal is defined to include operation beyond the original license term of 40 years, requiring a program for understanding and managing of plant SSCs.

The first license for a large plant (>400 MWe) will not expire until the year 2007. However, to allow for the long lead times required for planning and construction, the utilities need to decide approximately 10 to 15 years before the end of the license period whether to request a plant LR or to construct replacement generating capacity. Utilities have defined a tentative schedule for several important steps in the LR process (see Section 5.0). Two representative LWRs were the subject of EPRI and DOE utility-sponsored pilot studies on plant life extension. These were the Surry PWR plant, Virginia Electric Power Company, (EPRI NP-6232M) and the Monticello BWR plant, Northern

States Power Company, (EPRI NP-5836M). Subsequently, a bidding process was used to select two lead plants to develop formal requests for LR. The two plants that were selected are the 175-MWe Yankee Rowe PWR and Monticello, a 545-MWe BWR. At a technical level, this project is to provide an initial evaluation of the effects of aging on commercial NPPs and establish the scope of the effort to extend the operating lifetime of these plants beyond their initial 40 years of licensed operation. The first submittal to the NRC is expected in 1991. If the NRC accepts the LR concept, many additional submittals for LR can be expected over the next decade (up to 45 by the year 2002).

### 1.3 ACCOMPLISHMENTS

From key results of NPAR aging studies, the NRC recognizes that age-related degradation, a process that occurs in all technologies, needs systematic inspection and assessment for NPP safety. Communication of the need to understand and manage aging has been a major contribution of the NPAR Program. For example, it has been acknowledged that "the focus of license renewal is mitigation and management of age-related degradation to ensure an adequate level of safety" (NRC November 1989, Session 1).

While applying a systematic, comprehensive research strategy adaptable to a broad range of SSCs, the NPAR laboratories have developed technical bases for identifying age-related degradation of SSCs and developed effective measures for understanding and managing aging in the SSCs. The current scope of the technical bases is defined in Sections 3.0 through 6.0 of this plan for potential users, both regulatory and industrial.

The NPAR program also has made progress on major topics, including the Shippingport Aging Evaluation, determining the risk significance of aging phenomena, recommendations for field inspections, maintenance, and residual-life investigations of major LWR components and structures.

Cited in Section 6.0 are numerous specific cases where NPAR Program results have been used in the regulatory process, including input to regulatory guides, resolution of generic safety issues, revisions to the technical specifications, and input to development of codes and standards. Technical results have been used to recommend improved maintenance techniques and surveillance methods.

Finally, while focusing on the need to address aging in the current licensing period, the NPAR Program has compiled technical information that contributes to the technical bases for the LR Rule, scheduled to be issued in April 1992.

#### 1.4 ORGANIZATION OF NPAR PROGRAM PLAN DOCUMENT

This revision to the program plan, NUREG 1144, Rev. 2, emphasizes the major NPAR Program elements: the hardware studies, the special topics, and the recent emergence of LR. Included for each element is status, accomplishments, and future plans. Organization of the plan is summarized below.

Section 2.0 describes the basis from which the NPAR program conducts aging research.

Section 3.0 contains an outline of the NPAR strategy, the systematic approach used in the program for assessing the effects of aging on plant safety-related SSCs. The criteria used to identify safety-related SSCs are discussed, as is the phased approach developed to study the effects of age-related degradation.

Section 4.0 reviews special topics relevant to the aging of NPPs that have been and are being investigated in the program.

Section 5.0 summarizes the issues involved in LR, from both the industry and regulatory perspectives. Plans and schedules for actions required to resolve the issues and complete the LR process for the first NPPs are also discussed.

Section 6.0 contains an overview of accomplishments of the NPAR Program in the regulatory process. A synopsis of results then follows. The discussion of utilization is divided into five categories: rulemaking, generic safety issues; maintenance and surveillance; inspection of safety systems and components; and codes and standards.

Section 7.0 contains an overview of the NPAR review and integration activities within NRC, with other government agencies, and with external institutions and organizations, domestic and foreign.

Section 8.0 describes the program coordination with other government agencies and external institutions and organizations, both domestic and foreign. It also contains a discussion of the purpose and activities of the NPAR Research Review Groups.

Section 9.0 contains the schedules and research requirements developed for the various NPAR activities and a discussion of them. The schedules include research activities supporting LR initiatives and the continuation of age-related confirmatory research.

Appendix A contains a brief overview of NPAR participants.

Appendix B discusses bases for understanding and managing aging in NPPs.

Appendix C contains further details, beyond the overview in Section 8.0, of ongoing programs related to aging and extended operation. The coordination

required between the NPAR investigations and other ongoing activities is discussed, with emphasis on the need to optimize the use of available resources.

Appendix D cites examples of alternative approaches for conducting NPAR Phase-II comprehensive assessments.

Appendix E lists publications of NRC contractors on various aspects of the Shippingport Station aging evaluation.

## 1.5 NUCLEAR PLANT AGING RESEARCH PLANNING INSTRUMENTS

- The following are major planning instruments used in the NPAR Program:
  - Multi-year - The NPAR Program Plan, NUREG-1144. This plan was first issued in (Morris and Vora) 1985. Revision 1 was issued in (Vora) 1987. This version, Revision 2, was issued in 1991.
  - Annual/Monthly/Periodic
    - Program assumptions/program briefs
    - Form 189s
    - Program reviews
    - Programmatic discussions at the individual laboratories and/or at NRC Headquarters
    - Monthly reports
    - Users' needs
    - Direction from Advisory Committee on Reactor Safeguards
    - Recommendations of Research Review Groups
    - Interactions with EPRI and DOE
    - Interactions with outside institutions and organizations, domestic and foreign.

## 2.0 BASIS FOR THE NUCLEAR PLANT AGING RESEARCH PROGRAM

The U.S. Nuclear Regulatory Commission (NRC) is responsible for ensuring that nuclear power plants are operated safely during their initial licensed period and continue to operate safely during extended operation. To help fulfill this responsibility, the NRC instituted the Nuclear Plant Aging Research (NPAR) program to understand how aging may affect safety-related systems, structures, and components (SSCs), to identify the measures available to manage age-related degradation, and to anticipate problems that may result from plant aging. A set of technical safety issues has been developed for the NPAR program to provide focus and direction for aging assessments of SSCs and to maintain safe plant operation. The basis for developing the technical safety issues is described, followed by an overview of their implementation.

### 2.1 BASIS FOR IDENTIFYING AND RESOLVING TECHNICAL SAFETY ISSUES

The Technical Integration Review Group for Aging and Life Extension (TIRGALEX) was established in 1986 by the Executive Director for Operations (EDO) to facilitate the planning and integration of NRC plant aging and license renewal/life extension activities. The initial objectives of TIRGALEX were to clearly define the technical safety and regulatory policy issues associated with plant aging and extended operation and develop a plan for resolving the issues in a timely, well-integrated, and effective manner.

The framework recommended by TIRGALEX for planning and integrating agency activities related to plant aging and license renewal (LR) was adopted in the NPAR Program. There has been an ongoing effort to coordinate the various NRC initiatives that are relevant to plant aging: first through TIRGALEX, subsequently through the Aging and Life Extension Coordinating Committee (ALEXCC), and currently through the NPAR Review Group.

### 2.2 TECHNICAL SAFETY ISSUES

A set of technical safety issues was developed by the TIRGALEX to provide focus and direction for the NPAR Program. These issues are based on operating experience, expert judgment, and risk significance. These technical safety issues must be addressed to ensure that safety levels are maintained as the service lives of the present generation of reactors increase.

- What SSCs are susceptible to aging effects that could adversely affect public health and safety? Which of these SSCs are maintained and are replaceable?
- What are the degradation processes of materials, components, and structures that could, if improperly maintained and/or not replaced, affect safety during normal design life and during extended life?



- How can operational readiness of aged SSCs be ensured during the 40-year design life and during extended life of a reactor?
- Are currently available nondestructive examination and test methods adequate to identify all relevant aging mechanisms before safety is affected? If not, what efforts are under way to improve the methods.
- What criteria are required to evaluate residual life of SSCs? What supporting evidence (data, analyses, inspections, etc.) will be needed?
- How should SSCs be selected for comprehensive aging assessments and residual-life evaluations? Which SSC(s) should be selected?
- How effective are current programs for mitigating aging (e.g., control of environment, maintenance, replacement, and repair)?
- What kinds of reliability assurance and maintenance programs will be needed to ensure operational readiness of aged safety systems and components?
- What additional changes will be needed in codes and standards to address aging? What schedule should be followed?

The technical safety issues were developed by first examining the nature of the aging process and then examining the potential role aging plays in plant safety, and the NRC's mission to address plant aging and extended operation and LR. The technical objectives of the NPAR program have been developed to address the technical safety issues. Together, the technical objectives and the technical safety issues provide the framework required for developing and guiding the individual aging assessments in the NPAR program.

### 2.3 IMPLEMENTATION

In accord with the TIRGALEX Integration Plan, the technical data currently developed in related projects and the regulatory needs identified by the Office of Nuclear Reactor Regulation (NRR) establish the priority of the NPAR Program activities. The Office of Nuclear Regulatory Research (RES) has the responsibility for carrying out the necessary research programs.

Hardware-oriented engineering research needed to resolve the issues related to aging is being conducted in the Division of Engineering under three programs. The NPAR Program for aging investigations on components and systems is being performed by the Electrical and Mechanical Engineering Branch of the Division of Engineering. The aging research program on the vessels, piping, steam generator, and nondestructive examination techniques is being performed by the Materials Engineering Branch of the Division of Engineering. The aging research program and civil structures is conducted by the Structural and Seismic Engineering Branch of the Division of Engineering.

As the principal activities in these research programs are completed, the research results are made available for use in the regulatory process. RES also makes use of research findings as they contribute to the developing regulatory criteria, guides and standards, and review procedures.

### 3.0 AGING ASSESSMENTS OF SPECIFIC COMPONENTS AND SYSTEMS UTILIZING THE NPAR PROGRAM STRATEGY

Aging assessments have been conducted under the Nuclear Plant Aging Research (NPAR) Program on approximately 40 categories of components and systems considered risk significant, including a cross section of representative electrical and mechanical components and systems. The selection of the components and systems has been based on criteria outlined in Section 3.1. After the selection process was completed, participating laboratories and contractors were assigned specific components and systems for aging assessments (see Figure 3.1). In all, five national laboratories and several private institutions and organizations are conducting aging assessments, which are listed in Appendix A.

The aging assessments are based on a phased approach identified as the NPAR Program Strategy, which is illustrated in Figure 3.2. The NPAR Program Strategy involves the following phases:

- Pre-Phase I - Prioritization of systems, structures, and components
- Phase I - Interim Aging Assessment
- Phase II - Comprehensive Aging Assessment
- Utilization of Research Results.

The processes for selecting components and systems and applying the NPAR Strategy to the aging assessments are discussed in the following sections. Approaches to understand and manage aging are discussed in Appendix B, including descriptions of key degradation mechanisms.

#### 3.1 PRE-PHASE I - PRIORITIZATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

Before Phase I of the NPAR Program was implemented, the specific structures, systems, and components (SSCs) to be studied were selected for in-depth engineering studies. The selection criteria were based on the potential risk contributed from failures of SSCs; experience obtained from operating plants; expert judgment on the tendency of age-related degradation; and user<sup>(a)</sup> needs. Studies by Vesely et al. (1983) and Davis et al. (1985) provided risk-based inputs to the selection process. User needs include resolving generic safety issues, revising regulatory documents, upgrading plant maintenance and surveillance, and developing technical bases for license renewal (LR) consideration.

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(a) In the U.S. Nuclear Regulatory Commission (NRC), the user is the Office of Nuclear Reactor Regulation (NRR).

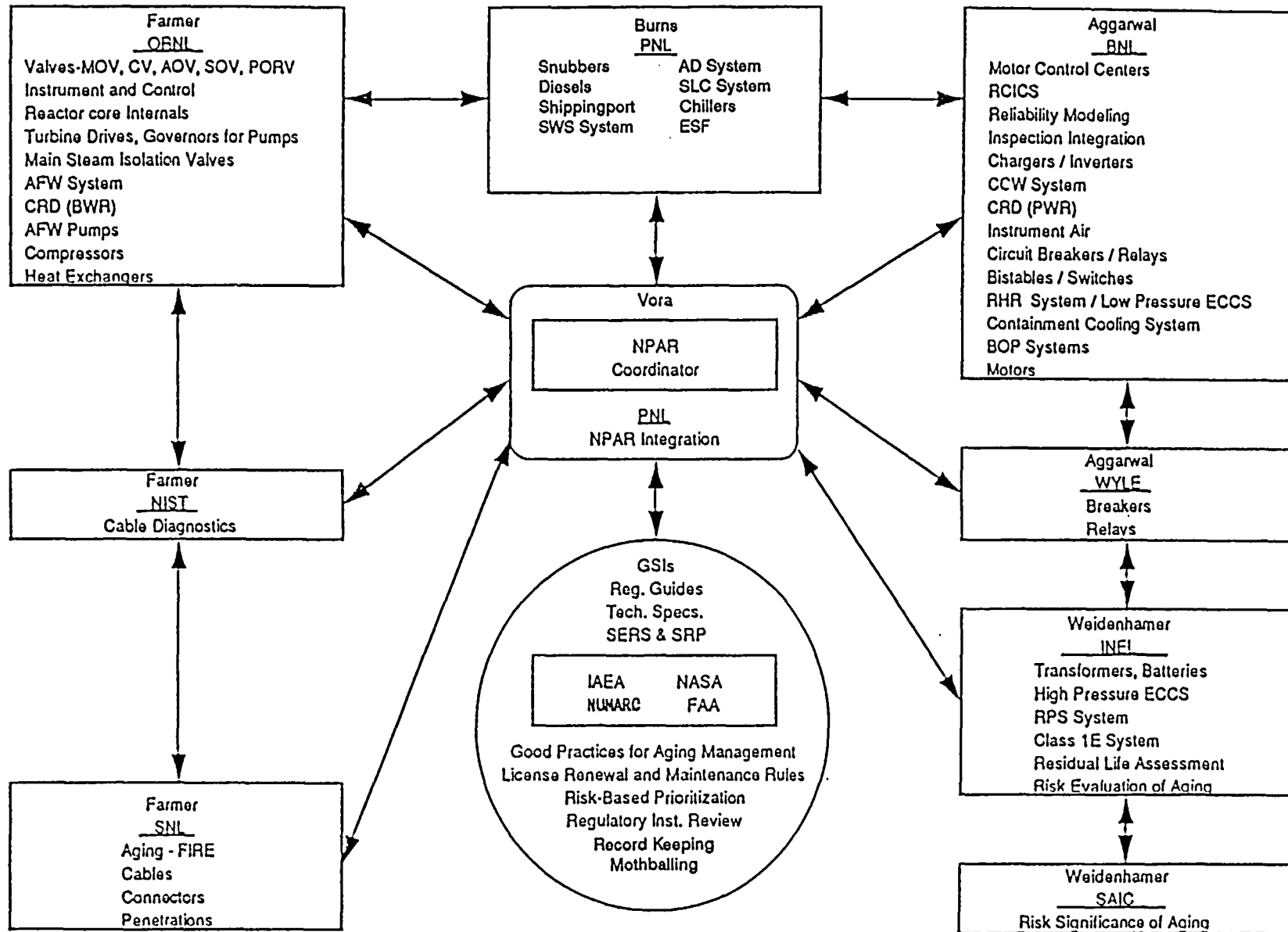
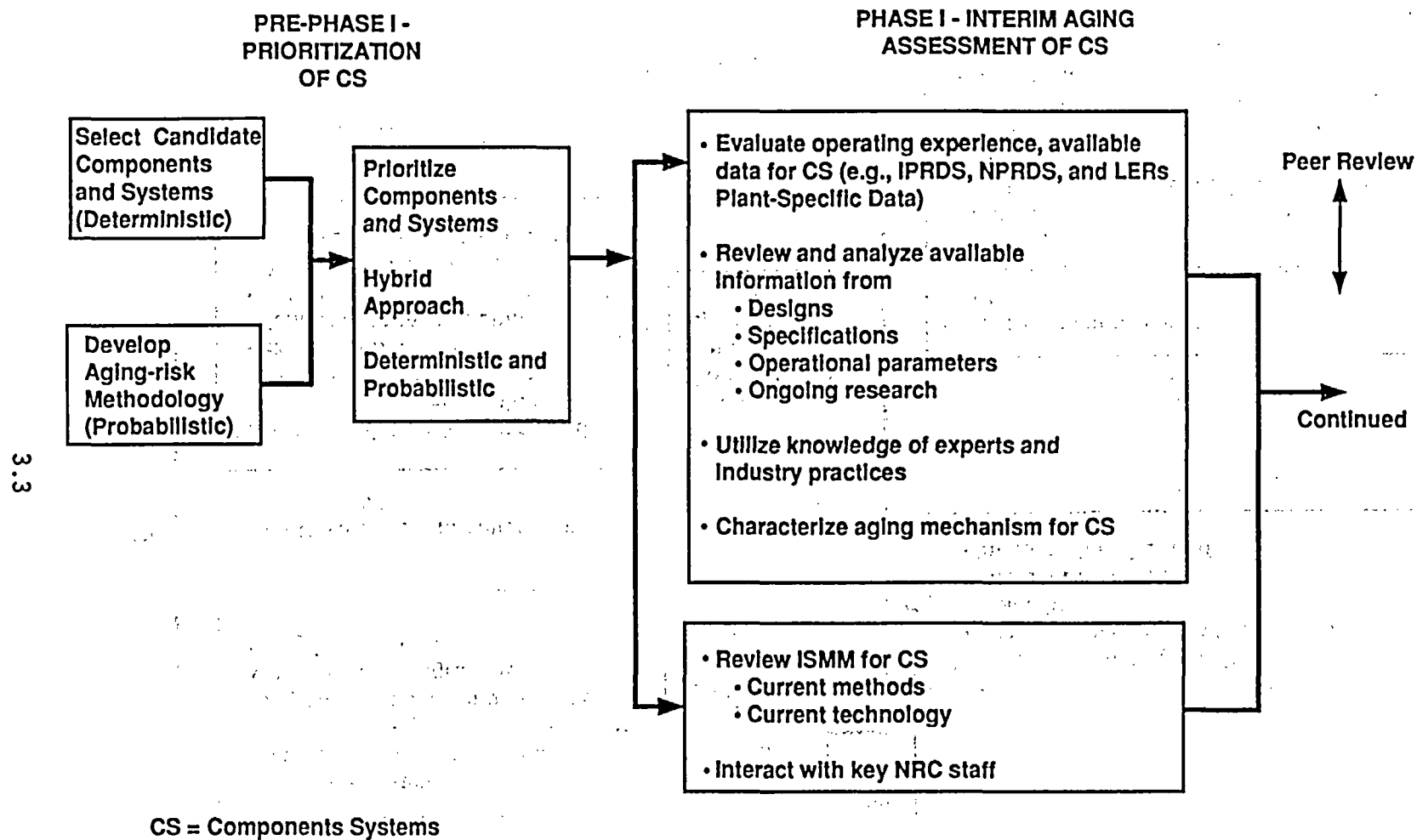


FIGURE 3.1. NPAR Structure

# NPAR Program Strategy



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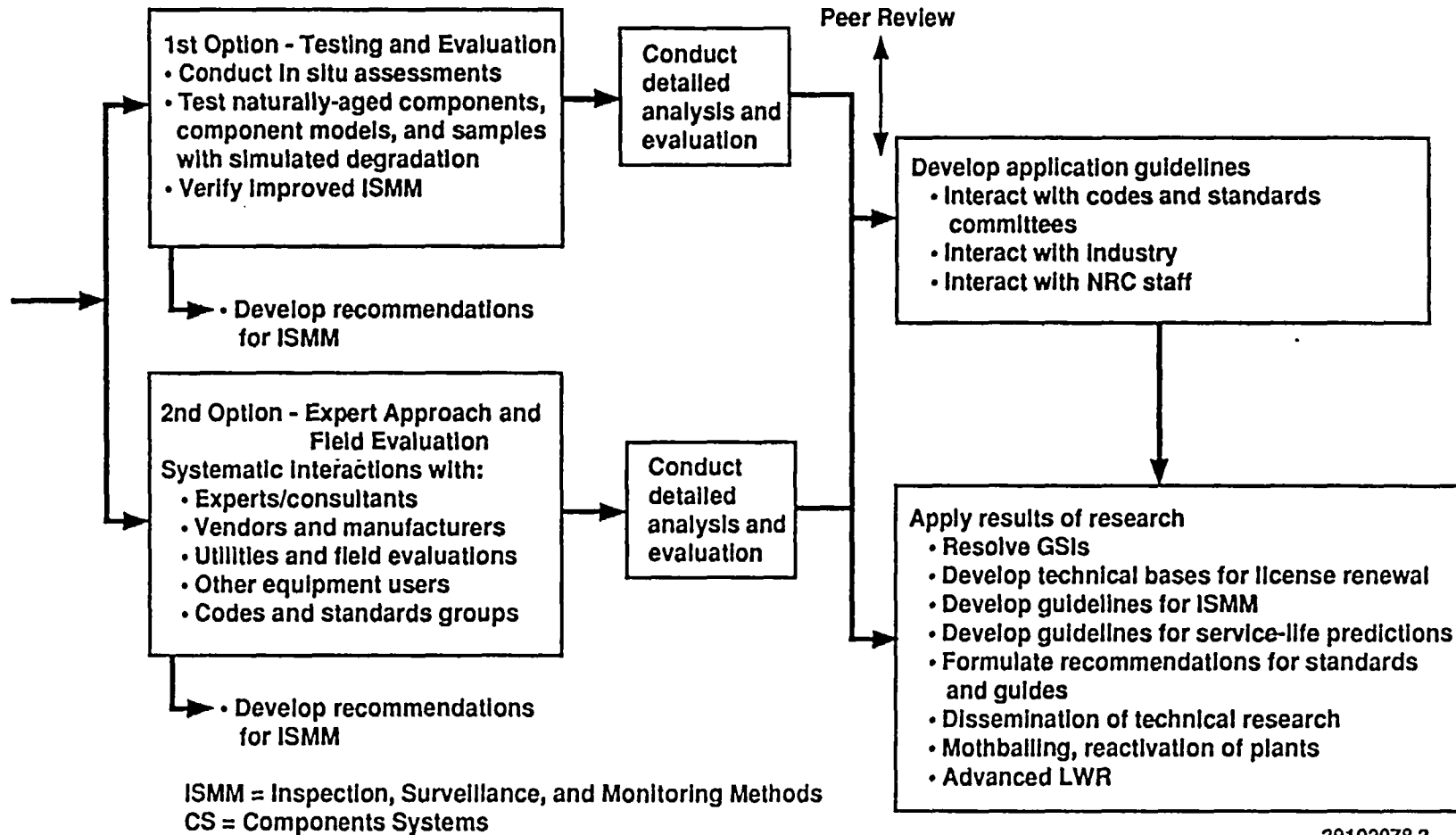
**FIGURE 3.2.** NPAR Program Strategy

# NPAR Program Strategy

## PHASE II - COMPREHENSIVE AGING ASSESSMENT OF CS - OPTIONAL APPROACHES

## UTILIZATION OF RESEARCH RESULTS

3.4



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FIGURE 3.2. (contd)

## **NPAR Program Strategy**

### **PHASE I ACTIVITIES**

- 1. Preliminary Identification of susceptibility of materials to age-related degradation.**
- 2. Identification of stressors and related environmental factors that could cause age-related degradation.**
- 3. Identification of degradation sites.**
- 4. Identification of failure modes and causes.**
- 5. Development of functional parameters to indicate present status and aging progression.**
- 6. Evaluation of current inspection, surveillance, and monitoring methods.**
- 7. Evaluation of current maintenance practices.**
- 8. Recommendations for Phase II studies.**

### **PHASE II ACTIVITIES**

- 1. In situ assessment of naturally-aged CS.**
- 2. Tests of naturally-aged components or tests of components with simulated aging.**
- 3. Laboratory or in-plant verification of methods of inspection, monitoring, and surveillance.**
- 4. Verification of methods of residual service-life evaluation.**
- 5. Development of recommendations for inspection or monitoring methods and periods.**
- 6. Recommendations for effective maintenance practices.**
- 7. Recommendations for acceptable methods to determine residual life of equipment.**

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**FIGURE 3.2. (contd)**

The major constituents of the SSCs were selected on the basis of the safety criterion that the release of fission products that could occur during an accident should be contained within the plant (Davis et al. 1985; Vesely 1987; Muscara and Serpan 1985; Vagins and Taboada 1985); these constituents include the pressure boundary components, containment, and supporting structures. Also included are components related to reactor control systems and reactor safety systems. The reactor internals are included because their failure may prevent control rod insertion or may cause fuel failure. Detailed evaluations of reactor pressure vessels, reactor coolant piping and safe ends, and steam generators are being performed in the program sponsored by the Materials Engineering Branch of the Division of Engineering.

The scope of NPAR aging assessments encompasses 1) pressurized-water reactors (PWRs) from the three U.S. nuclear steam supply system vendors, 2) boiling-water reactors (BWRs), 3) plants with numerous variations in design, applications, and suppliers, and 4) operation and maintenance with differing practices and philosophies. The NPAR Program includes studies of selected electrical and mechanical components and representative safety systems and support systems, special topics studies, and the utilization of results from aging assessments.

NOTE: It is not the intent of the NPAR Program to conduct in-depth engineering evaluations of aging and defect characterization, and methods for inspection, surveillance, and monitoring methods and mitigation practices of all significant plant elements. It is the industry's responsibility to characterize and evaluate their own plant SSCs to ensure their operational safety as the plants advance in age.

Table 3.1 contains a listing of components that have been identified to have aging-related impacts on plant safety and support systems and their availability and safety margins. Current status of work on the study of the aging of these components is also noted.

Table 3.2 provides the research status of systems in nuclear power plants (NPPs) that are of current interest. These systems are considered important for accident prevention or mitigation.

### 3.2 PHASE I - INTERIM AGING ASSESSMENT

After the components and systems are selected, the first step is to establish a boundary to define what is to be included in the component or system under consideration and also to identify important interfaces between the component or system to be investigated and other adjacent components or systems. The Phase-I assessment is based on information from public and private databases, vendor information, open literature, utility sources, operating experience, and expert opinion. The information includes the identification of failure modes, preliminary identification of failures caused by age-related degradation, and reviews of inspection, surveillance, and monitoring methods



**TABLE 3.1. Components of Current Interest in the NPAR Program and Their Completion Schedule**

<u>Topic</u>	<u>Laboratory</u>	<u>Schedule</u>
Motor operated valves	ORNL	Complete in FY-91
Check valves	ORNL	Complete in FY-91
Solenoid valves	ORNL	Complete in FY-91
Air operated valves	ORNL	Initiate Phase 1 in FY-91
Auxiliary feedwater pumps	ORNL	Complete in FY-91
Small electric motors	ORNL	Completed in FY-88
Large electric motors	BNL	Initiate Phase 1 in FY-92
Chargers/inverters	BNL	Completed in FY-90
Batteries	INEL	Completed in FY-90
Power operated relief valves	ORNL	Completed in FY-89
Snubbers	PNL	Complete Phase 2 in FY-91
Circuit breakers/relays	BNL, Wyle	Complete Phase 2 in FY-91
Electrical penetrations	SNL	Complete Phase 1 in FY-91
Connectors, terminal blocks	SNL	Initiate Phase 1 in FY-91
Chillers	PNL	Initiate Phase 1 in FY-91
Cables	SNL	Complete Phase 2 in FY-91
Diesel generators	PNL	Phase 2 completed in FY-89
Transformers	INEL	Complete Phase 1 in FY-91
Heat exchangers	ORNL	Complete Phase 1 in FY-91
Compressors	ORNL	Phase 1 completed in FY-90
Bistables/switches	BNL	Initiate Phase 1 in FY-91
Main steam isolation valves	ORNL	Initiate Phase 1 in FY-91
Accumulators		No initiative
Surge arrestors		No initiative
Isolation condensers (BWR)		No initiative
Purge and vent valves		No initiative
Safety relief valves		No initiative
Service water and component cooling water pumps		No initiative

**TABLE 3.2. Systems of Current Interest in the NPAR Program and Their Completion Schedule**

<u>Topic</u>	<u>Laboratory</u>	<u>Schedule</u>
High pressure emergency core cooling system	INEL	Complete Phase 1 in FY-91
RHR/Low pressure emergency core cooling system	BNL	Complete Phase 2 in FY-91
Service water	PNL	Phase 2 completed in FY-90
Component cooling water	BNL	Complete Phase 2 in FY-92
Reactor protection	INEL	Complete Phase 2 in FY-91
Class 1E electric distribution	INEL	Complete Phase 2 in FY-91
Auxiliary feed water	ORNL	Initiate Phase 1 in FY-91
Control rod drive, PWR (W)	BNL	Phase 1 completed in FY-90
Control rod drive, PWR (B&W, CE)	BNL	Complete Phase 1 in FY-92
Control rod drive, BWR	ORNL	Complete Phase 1 in FY-91
Motor control centers	BNL	Completed in FY-89
Instrument air	BNL	Complete Phase 2 in FY-92
Containment cooling	BNL	Complete Phase 1 in FY-91
Engineered Safety Features	PNL	Initiate Phase 1 in FY-91
Instrument and control	ORNL	Complete Phase 1 in FY-92
Automatic depressurization (BWR)	PNL	Complete Pre-Phase 1 in FY-91
Standby liquid control (BWR)	PNL	Complete Phase 1 in FY-91
Core internals	ORNL	Initiate Phase 1 in FY-91
Turbine main generator and controls	ORNL	Initiate Phase 1 in FY-91
Containment isolation		No initiative
Recirculation pump trip actuation instrumentation (BWR)		No initiative
Reactor core isolation cooling		No initiative

(ISMM), including manufacturer-recommended surveillance and maintenance practices. Performance parameters or condition indicators potentially useful in detecting degradation are identified, and preliminary recommendations are made regarding improved ISMM and maintenance practices. Recommendations are also developed to identify the detailed engineering tests and analyses conducted in Phase II.

A key outcome of the Phase-I investigation is a decision regarding the need for a Phase-II assessment on specific SSCs. The following sections summarize the activities that comprise the Phase-I interim investigation.

### 3.2.1 Consultations with Nuclear Regulatory Commission Staff

An important Phase-I consideration is to identify where similar aging research or related regulatory initiatives are active in the NRC while avoiding duplication. Aging research should potentially augment the value of the related research and regulatory initiatives. For example, an NRR study on service water system (SWS) biological fouling provided important input to an NPAR SWS aging investigation. Identifying related regulatory initiatives, such as the resolution of generic safety issues or needed revisions to regulatory guides or technical specifications, can determine how NPAR aging research may contribute to resolving regulatory needs.

### 3.2.2 Review of Design Information and Applications

During the preliminary research, design information and applications are reviewed as follows:

1. Design and Specifications. The first of the three elements of the Phase-I assessment begins with a review of the design data and specifications for the hardware being studied. This includes such items as nonproprietary design documents, final safety analysis reports, operating and maintenance manuals, and product literature. Additional sources of information are also reviewed. These include vendor surveys, utility contacts, published reports, and expert opinion.
2. Materials. An important aspect of Phase I is to identify all susceptibilities of significant materials that comprise the hardware under review. The susceptibilities of specific materials and parts to age-related degradation are evaluated.
3. Operating and Environmental Stressors. The degradation of the SSCs involves time-dependent phenomena and, among other things, depends on operating environment and operating history. The environmental stressors considered include temperature, radiation, chemicals, contaminants, atmospheric conditions, and humidity. The environmental conditions considered include operational parameters and environmental conditions that prevail during other periods, such as during testing, shutdowns, storage periods, accident conditions, and post-accident situations.

The operating history assessment includes the thermal, mechanical, and electrical stressors that SSCs experience during their operating lifetimes. Normal operating conditions, anticipated transients, off-normal conditions, and accident and post-accident conditions are studied to determine the influence and effects of stressors and

environment on degradation processes. Typical examples of electrical stressors include slow-switching transients, fast transients of the lightning variety, and low frequency 50- to 60-Hz signals, which can occur singly or in various combinations. Examples of mechanical stressors are static loading stresses, dynamic loading stresses, and seismic and vibrational stresses.

4. Performance Requirements and Condition or Functional Indicators. Performance requirements are reviewed to assess whether aging degrades the ability of SSCs to perform required safety functions during normal, abnormal, and accident conditions. Here it may be possible to identify condition or functional indicators. These consist of indicators that are practical to monitor and that provide cost-effective means to identify and manage degradation. Finally, ongoing research is reviewed, and applicable results are included in the assessment of hardware under study.

This review and analysis of materials, designs and specifications, stressors and environment, and operational parameters are performed on all components and systems selected for the comprehensive aging assessment.

### 3.2.3 Survey of Operating Experience and Failure Evaluation

Another activity of the Phase-I assessment is a critical survey of the documented operating experience obtained on SSCs being evaluated. This review provides information on the failure rates and reliability that can be expected, and the aging-related failure modes and causes that have been experienced. The sources of this information include data from the in-plant reliability data system (IPRDS) sponsored by the NRC and the nuclear plant reliability data system (NPRDS) managed by the Institute of Nuclear Power Operation. Other sources include Licensee Event Reports (LERs), Nuclear Plant Experience, Plant Maintenance Records, and Inservice Inspection Reports.

From the critical survey, hardware failures are evaluated to identify the following:

- Failure Mechanisms. These are established through the process of identifying dominant stressors; studying materials and designs of components and parts; reviewing service environments and applications; and then evaluating the nature of, and the factors contributing to, aging and failures.
- Failure Modes. The indicators of failures (for example, leaks, fatigue failures, radiation, thermal embrittlement of materials, and voltage collapse or disturbance in current signature) are assessed and critical age-related failure modes are identified.
- Failure Causes. The conditions of design, manufacture, service environments, and applications that may lead to failures are determined.

### 3.2.4 Review and Evaluation of Inspection, Surveillance, and Monitoring

The third element in the Phase-I investigation is the review of the inspection, surveillance, and monitoring methods (ISMM). This element also involves a review of artificial or accelerated aging techniques used to qualify hardware for end-of-life performance.

Existing methods for ISMM are evaluated to determine those methods likely to be effective in detecting aging at an incipient stage before loss of safety function. Also it is important that the methods not be unduly expensive to implement and not result in unacceptable levels of occupational exposure. Surveillance and monitoring methods being evaluated include periodic inspections, both visual and instrument-aided, and on-line instrumented techniques. The evaluation seeks to identify performance parameters and functional indicators that represent the capability of equipment and are useful for managing aging. Key considerations are to monitor the selected parameters and indicators at operating plants for reasonable costs.

Artificial or accelerated aging techniques may be reviewed and compared to data available from naturally-aged hardware to demonstrate the applicability of current practices.

### 3.2.5 Interim Aging Assessment and Recommendations

The result of a Phase-I evaluation of SSCs is an interim aging assessment and an evaluation of the safety significance of the probable failure modes (Wu 1989). An interim evaluation of current ISMM technology is given, and potential performance parameters and condition indicators are developed. Interim recommendations are made for Phase-II studies based on the results and reviews of research activities completed in Phase I. The results of the Phase-I study are issued in a technical progress report or a NUREG/CR report. Generally, NUREG/CR reports are subject to peer review before publication.

Continuation to Phase-II activities, for a given component or system, is deemed unnecessary if 1) an adequate database and experience exists within the industry; 2) industry-sponsored programs adequately address the research needs; and 3) resources need to be directed to other research activities.

## 3.3 PHASE II - COMPREHENSIVE AGING ASSESSMENT

Phase-II assessments of SSCs usually involve some combination of one or more of the following: 1) tests of naturally-aged equipment or equipment with simulated degradation; 2) laboratory or in-plant verification of methods for ISMM; 3) aging assessments by expert panels; 4) development of recommendations for inspection or monitoring techniques; 5) verification of methods for evaluating residual service lifetime; 6) identification of effective maintenance practices; 7) in-situ examination and data gathering for operating equipment; 8) verification of failure causes using results from in-situ and postservice examinations.

When fully conducted, the Phase-II assessment involves several in-depth assessments, which include validating advanced ISMM through laboratory and field testing of samples and validating accelerated aging techniques. It also may include developing models to simulate degradation, in-situ aging assessment, and testing of naturally-aged equipment from operating NPPs.

In-depth laboratory investigation of selected naturally-aged components has been an element of NPAR Phase II for certain categories of equipment such as valves, motors, and battery chargers/inverters. However, detailed laboratory investigations on large components such as emergency diesel generators (EDGs) are not practical at current NPAR funding levels. Therefore, the Phase-II assessment of such components has involved and will continue to involve extensive use of experts and coordination with utilities and other industry sources to identify aging characteristics. Budget limitations have required developing innovative approaches to investigations of aging mechanisms without reliance on in-depth laboratory studies and studies on naturally-aged components. Alternative approaches to comprehensive aging assessments that have been developed and applied in NPAR studies are described in Appendix D.

### 3.3.1 Review and Verification of Improved Inspection, Surveillance, and Monitoring Methods

The Phase-II research of ISMM involves reviewing advanced methods and technology for each category of components and systems under study. In this phase, advanced techniques and technologies, either in use or under development, are investigated. When available, the sources of information technology both within and outside the nuclear industry are used. The sources outside the nuclear industry include fossil plants, the petrochemical industry, the aerospace industry, various branches of the Department of Defense, and other government agencies. The practical feasibility of applying these technologies to nuclear plant components is also explored.

Laboratory and field application and verification tests of ISMM candidate technologies are carried out. The objective of the tests is to demonstrate that methods are appropriate to follow the dynamics of the performance parameters and functional indicators of interest; methods have adequate selectivity (will not give false indications) and sensitivity (will detect in the incipient stage); and suitable acceptance/rejection criteria are available so that maintenance needs can be correctly identified.

Some laboratory tests could involve simulating defects of varying degrees of propagation in prototype hardware to determine sensitivity and detection criteria. Various defect and environment combinations could be used to determine selectivity. These laboratory tests could be conducted to verify that the methods are applicable for in-situ use at power plants. Field tests are recommended at cooperating utilities to confirm the laboratory results, to provide information about the frequency and method of data collection and analysis, and to estimate cost effectiveness and practicality of application.

### 3.3.2 Testing of Naturally-Aged Components

A second element in the Phase-II assessment is examining and testing naturally-aged components that have had extended service in NPPs. Obtaining and testing naturally-aged components is sometimes difficult or costly; however, the information obtained from the testing is valuable to quantify the effects of aging and to determine whether adequate safety margins exist to ensure the operational readiness of naturally-aged components and systems that remain in service.

Equipment that has experienced significant operating and environmental stressors was obtained from the decommissioned Shippingport PWR. Some examinations of the naturally-aged Shippingport equipment were conducted in-situ. Other Shippingport components and specimens were shipped to participating laboratories for aging evaluations. Results from the Shippingport aging studies that have been reported are referred to in Section 4.3 Appendix E.

In-situ monitoring of operating equipment at LWRs is recommended to gain an understanding of the interaction between age-related defect characterization and inspection, surveillance, and maintenance. Also, aging investigations are sometimes undertaken on equipment that has failed during operation. Determining the root cause of the failure is the key consideration.

### 3.4 UTILIZATION OF RESEARCH RESULTS

With the completion of the aging assessment of SSCs, a technical basis becomes available for use in developing and refining the regulatory process. Examples of possible uses include implementing improved ISMM; modifying present codes and standards; resolving generic safety issues; and developing guidelines and review procedures for LR.

In the NPAR Program Strategy (Figure 3.2), the research performed for the Phase-I and Phase-II assessments leads to developing application guidelines. It may also provide a systematic collection of historical baseline data and trending information for evaluating component and system aging effects. The specific areas of application, shown in Figure 3.2, provide highlights of the end uses of the NPAR Program. An important part of the application guideline phase is the technical integration of the results obtained by NPAR and other major programs. In developing guidelines, the NPAR staff work with all the NRC offices involved in programs relevant to nuclear plant aging and LR, with codes and standards committees, and with industry groups.

#### 3.4.1 Resolution of Generic Safety Issues

The NRC report, NUREG-0933 (Emrit et al. 1991) contains a recommended priority list to assist in the timely and efficient resolution of generic safety issues. The NPAR Program generates guidelines, develops criteria, and supports resolution of the generic safety issues. The generic safety issues that would directly benefit from the NPAR Program results are listed in Table 3.3.

**TABLE 3.3. Generic Safety Issues,<sup>(a)</sup> with Elements of Aging**

<u>Issue Number</u>	<u>Title</u>
15	Radiation Effects on Reactor Vessel Supports (Revised)
23	Reactor Coolant Pump Seal Failures
51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems
87	Failure of HPCI Steam Line Without Isolation
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers
130	Essential Service Water Pump Failures
142	Leakage Through Electrical Isolators in Instrument Circuits
143	Availability of Chilled Water System and Room Cooling
153	Loss of Essential Service Water in LWR's
B-55	Improve Reliability of Target Rack Safety Relief Valves
B-56	Diesel Reliability
S11A	License Renewal Rulemaking
S17	Electric Power Reliability

(a) From NUREG-0933 (Emrit, et al. 1991).

### 3.4.2 Considerations for License Renewal

An important objective of the NPAR Program is to identify and resolve the technical safety issues involved in requests for LR of nuclear power plants. An end product of NPAR initiatives will be guidance or recommendations to NRC users on subjects such as revisions to ISMM methods, residual lifetimes of major components, key technical information required in applications for LR, ensuring continued safe operation of plants, and providing support to the LR rulemaking process. This topic is discussed in Section 5.0.

### 3.4.3 Guidelines for Inspection, Surveillance, Monitoring Methods, and Maintenance

A principal element in the engineering evaluation of age-related degradation is evaluating methods used in ISMM of NPP systems and components. A review is made of methods currently in use in Phase I. Also, an evaluation is provided of their effectiveness in detecting age-related degradation at an



incipient stage before a loss of safety function. Advanced techniques are also reviewed, and may be tested in Phase II.

The results of the NPAR aging assessments are also utilized to evaluate the role of maintenance in mitigating age-related degradation and developing guidelines for revised or preferred maintenance practices. This work consists of the following activities: reviewing current practices and procedures; reviewing vendors' recommendations; evaluating the merits of performing preventive, corrective, or predictive maintenance; identifying failures caused by maintenance procedures; and developing recommendations for a preferred maintenance approach.

#### 3.4.4 Guidelines for Service-Life Predictions

Aging and residual life of major LWR components is evaluated in the NPAR Program. The current methods for predicting the service life of major mechanical components and structures are reviewed to develop an approach for life-time evaluation. The resources required to inspect and monitor components and structures are considered to determine whether technically acceptable methods for predicting service life could be substituted. The information generated in this research has two principal objectives. One is to assist in developing criteria that ensure that the degradation of major components does not impair safe plant operation. The second is to generate a technical basis for establishing criteria and developing guidelines to be used in license review procedures for LR. Although these objectives complement the objectives of ongoing industry-sponsored pilot projects on LR, they focus only on safety aspects.

The approach to the residual-life assessment involves identifying and prioritizing major components with respect to safe plant operation. This is followed by an initial effort to determine the life-limiting processes for each of the major components, whereby degradation sites and failure modes during normal operation and accident conditions are identified.

The residual-life assessment also includes current and potential methods for ISMM. In the utilization phase, the work is focused on integrating currently available technical information relevant to aging during current operations and during LR. Further description of residual-life assessments of major components is addressed in Section 4.4.

#### 3.4.5 Recommendations for Standards and Guides

The NPAR Program develops recommendations for revising relevant industry codes and standards for continuing the operation of aged plants. The NPAR Program also provides input to technical bases for preparing NRC regulatory guides, review procedures for the continued operation of NPPs, and for LR considerations.

#### 3.4.6 Dissemination of Technical Results

The research information developed in NPAR is disseminated by preparing technical papers (for example, journal articles) and reports (for example, NUREG/CRs) and by sponsoring workshops, training seminars, society presentations, symposia, information exchange programs, and international (IAEA) initiatives. To investigate approaches to make the NPAR technical bases more accessible, two EDG reports (one sponsored by NPAR, one sponsored by DOE) were compiled on computer disks and were cross-referenced using a Hypertext approach. Wider use of the approach to improve accessibility of this database is being considered.

#### 3.4.7 Innovative Materials and Design

Another element in the NPAR Program Strategy (Figure 3.2) is the identification of needs for innovative materials and designs. Here recommendations may be provided (and it is up to the industry to implement them) to evaluate design changes to existing SSCs that would make them less susceptible to aging. Innovative materials and designs could also find end uses in the other NRC programs (such as equipment qualification and advanced LWR designs).

## 4.0 SPECIAL TOPICS

The special topics, listed in Table 4.1 and reviewed in this section, constitute a major element of the Nuclear Plant Aging Research (NPAR) Program. Most of the research on these topics interacts with the research on the hardware-oriented initiatives discussed in Section 3.0. For example, the Shippingport aging evaluation has provided naturally-aged components to several of the systems and components studies. Most of the remaining topics derive inputs from the hardware studies; for example, specific needs for

**TABLE 4.1. Special Topics of Current Interest in the NPAR Program and Their Completion Schedule**

Topic	Laboratory	Schedule
Risk evaluations of significant aging effects	SAIC	Complete Phase 1 in FY-90
Risk-based component prioritization and selection	PNL	Complete Phase 1 in FY-90
Shippingport aging assessment	PNL	Complete in FY-90
Aging assessment and mitigation of major LWR components and structures	INEL	Complete Phase 1 in FY-90
Development of technical bases for:		
• Regulatory guides	PNL	Complete Phase 1 in FY-91
• Standard review plans	PNL	Complete Phase 1 in FY-91
• Criteria for license renewal	PNL	Complete Phase 1 in FY-91
• Maintenance rulemaking	PNL	Continuing
• Manual for aging management	PNL	Complete Phase 1 in FY-91
Technical specifications from aging perspective	PNL	Complete Phase 1 in FY-91
Guidance for recordkeeping	PNL	Complete Phase 1 in FY-91
Inspection integration	BNL	Complete Phase 1 in FY-91
Reliability modeling of component aging	BNL	Complete Phase 1 in FY-91
Review of regulatory instruments	PNL	Complete Phase 2 in FY-91
Review of industry technical reports for license renewal	(All)	Complete in FY-91
Aging Effects in Balance-of-Plant Systems	BNL	On Hold
Maintenance to manage aging	PNL	Complete Phase 1 in FY-90
Aging degradation of cables	SNL	Complete in FY-90
Fire vulnerability of aged electrical components	SNL	Complete Phase 1 in FY-90
Aging effects in mothballed PNL plants	PNL	Planned for FY-92

changes to the technical specifications and nuclear power plant (NPP) record-keeping are identified in part from the NPAR aging studies identified in Section 3.0. The special topics also are providing important input to NRC license renewal (LR) initiatives.

The scope and status of the special topics identified in Table 4.1 are summarized in the following sections.

#### 4.1 RISK EVALUATIONS OF SIGNIFICANT AGING EFFECTS

Objective: To develop and demonstrate an age-dependent PRA-based methodology for prioritizing the risk contributions to and maintenance importances of aged components.

Contractor: Science Applications International Corporation (SAIC)

Methodology: A methodology has been developed that takes into consideration the aging contributions to core melt frequency and to other risk results by a Taylor expansion of the basic risk equations. This approach allows current probabilistic risk assessments (PRAs) to be used to determine the risk importance that can then be combined with aging effects on components and structures, which are determined using separate aging models. Using this approach, PRAs do not have to be recalculated to incorporate the aging effects; instead, current PRA results can be combined with aging effects of a component and structural level to obtain the core melt frequency increases and other risk result increases caused by aging. In this methodology, multiple aging effects are incorporated, aging contributors are prioritized in detail, and maintenance programs can be quantified for their risk effectiveness in controlling aging effects. Uncertainty evaluations are also included.

Status: The methodology has been tested and demonstrated. A NUREG/CR has been issued (Vesely, Kurth, and Scalzo 1990). Software programs have been initially developed for applications of the methodology. Data analysis programs have also been developed to analyze data for aging effects that apply to the risk evaluations to determine the risk implications of age-related degradation.

Results: Three recent PRAs have been used to evaluate the risk effectiveness of test and maintenance programs using the linear aging model. Analyses have also been conducted to support regulatory analyses for LR rulemaking.

Plans: Plans are to extend the methodology to include more comprehensive models of aging effects, test procedures, and maintenance programs. Further plant-specific data are to be analyzed, and production schemes are to be developed. Possible applications to assist regulatory decision-making will also be investigated.

## 4.2 RISK-BASED COMPONENT PRIORITIZATION AND SELECTION

**Objective:** To develop and demonstrate methodologies and procedures for implementing a hybrid (PRA-based and deterministic) prioritization of aged components and structures.

**Contractor:** Pacific Northwest Laboratory (PNL)

**Methodology:** The PNL hybrid approach selects and prioritizes all important-to-safety components based on risk contribution resulting from component risk importance and aging effects. The hybrid approach accomplishes this by combining the active components selected by an age-dependent PRA-based methodology with passive components and structures (SC) not included in the PRA. PNL's approach will seek to incorporate the SC not included in the PRA by using the results from the PRA as input to an expert-panel deterministic method, the Analytical Hierarchy Process (AHP). In this approach, the risk importance for active components from the PRA are used to provide benchmarks for the AHP values for these components. The relative risk importance of the passive components and structures would then be derived from their hierarchy relative to the active components by the expert panel. The effects of aging on the failure probability of each passive SC will then be calculated using inputs on failure rate, effective test interval, and effective renewal/replacement interval, as was previously done for the PRA-based active components. The resultant aging failure probability of each SC will be combined with its risk importance to produce the resultant risk contribution from age-related degradation.

**Status:** The PRA-based component prioritization methodology incorporating the effects of multiple component aging was developed. This PRA-based methodology was used to prioritize active components using two NUREG-1150 PRAs, one each for a PWR and BWR. From the prioritization, high-priority components were identified and selected for improved maintenance, which was implemented through reductions in the inspection and renewal intervals. A report on this effort<sup>(a)</sup> is undergoing peer review at NRC before its anticipated publication by the end of 1991.

**Results:** The PRA-based studies provided two important perspectives on aging effects. The first is that aging of plant components may result in a significant increase to plant risk; the amount of increase may be plant-specific. The results of incorporating the effects of multiple component aging demonstrated that aging may be viewed as an important common cause of component failure. Integrating single and multiple components in a prioritized list of components also demonstrated the importance of developing more comprehensive PRAs to incorporate all important contributors to plant risk.

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(a) Levy, I. S., W. E. Vesely. Draft September 28, 1990. PRA-Based Prioritization of Risk Contributions and Maintenance Importance for Aged Active Components. NUREG/CR-5587, Pacific Northwest Laboratory, Richland, Washington.

Plans: Completion of the hybrid component selection methodology is expected by the end of 1992.

#### 4.3 SHIPPINGPORT AGING ASSESSMENT

Objective: To use data and naturally-aged components and specimens from the decommissioned Shippingport pressurized-water reactor (PWR) to evaluate aging effects, resulting from reactor service, on components and systems being investigated in the NPAR Program and by other NRC organizations.

Contractor: Pacific Northwest Laboratory (PNL)

Methodology: As the Shippingport PWR was decommissioned, the aging assessment was conducted by 1) evaluating the relevance of Shippingport to operating reactors; 2) in-situ measurements on circuits and materials; 3) removing samples from selected components; 4) acquiring components and systems for use in studies at participating laboratories;<sup>(a)</sup> and 5) acquiring operating data, manuals, etc., to accompany equipment items that were removed for investigation by NPAR and other NRC organizations.

Status: Over 200 components (e.g., battery chargers, valves, a motor control center) and specimens (e.g., from neutron shield tank, concrete, cast stainless steel components, piping) were removed and sent to participating laboratories; all buildings and equipment have been removed and the decommissioning is complete; PNL continues to provide assistance with data and information on the components and specimens that have been acquired.

Results: Data from examining the Shippingport components and specimens are providing characteristics of thermal and radiation embrittlement (by ANL), aging characteristics of inverters/battery chargers (by BNL), check valves and motor operated valves (by ORNL) and radiological assessments from coolant system samples (by PNL). A cable sizing problem on a motor operated valve was identified (by INEL), resulting in issuance of NRC Information Notice No. 89-11.

Plans: Investigations are continuing on components and specimens from Shippingport; PNL will continue to provide records support while the investigations are in progress. Attention will be given to proper disposal of Shippingport equipment used in the investigations.

- 
- (a) Elements and participants in the Shippingport aging evaluation:
- |                        |                            |
|------------------------|----------------------------|
| - Coordination         | PNL                        |
| - In situ assessments  | ANL, INEL, PNL             |
| - Components           | BNL, INEL, NIST, ORNL, PNL |
| - Samples of materials | ANL.                       |

#### 4.4 AGING ASSESSMENT AND MITIGATION OF MAJOR LWR COMPONENTS AND STRUCTURES

Objective: To develop an understanding of the aging effects of major light-water reactor (LWR) components and structures so that the effects can be effectively managed. The major emphasis of this project is on integrating, evaluating, and updating the research results from related programs, including those sponsored by the NRC and industry. The five specific objectives of this project are to 1) identify and evaluate technical safety issues related to plant aging and LR; 2) evaluate and define surveillance and mitigation methods; 3) recommend revisions of codes and standards to support aging management and LR; 4) identify additional research projects needed to resolve the technical safety issues related to plant aging and LR; and 5) develop and evaluate life-assessment models and procedures for the major LWR components and structures.

Contractor: Idaho National Engineering Laboratory (INEL)

Methodology: The project consists of integrating, evaluating, and updating all the relevant technical information on degradation mechanisms, nondestructive inspection techniques, and monitoring methods relevant to major LWR components and structures. The major sources of information include reports from completed and ongoing NRC and industry research programs; NRC Information Notices, Bulletins, and Generic Letters; and the Nuclear Power Experience database. Contributions from several technical experts from industry, universities, and laboratories are sought to prepare the project reports. Several other experts are sought to critically review the reports. The project results are being used to develop and modify regulatory guides, technical specifications, Standard Review Plans, and ASME Section XI Codes and Standards related to aging and LR. A five-step approach is used to accomplish the project objectives:

1. Identify and prioritize major components and structures according to relevance to plant safety.
2. Identify for each component, degradation sites, mechanisms and stressors; identify potential failure modes; and then evaluate current inservice inspection methods.
3. Assess advanced inspection, surveillance, and monitoring methods that can be used to detect, size, and trend age-related degradation. Evaluate current and emerging methods to mitigate aging damage.
4. Develop life-assessment procedures for LWR components and structures.
5. Support the development of technical criteria for LR.

Status: Qualitative aging assessments of 22 major components including primary pressure boundary components, containments, emergency diesel generators, and cables and connectors have been completed. This assessment marks the completion of the first two steps. The remaining three steps are

partially completed. For the third step, evaluation methods for materials properties, monitoring techniques for acoustic emissions, and methods for monitoring fatigue are being evaluated. As part of the fourth step, draft reports on the life-assessment procedures for PWR reactor pressure vessels, LWR metal containments, PWR steam generator tubes, and LWR cast stainless steel components are published. In the fifth step, the project personnel provide input to the reports, which are being prepared by PNL for the technical support of the regulatory activities related to aging and LR.

Results: Qualitative assessment has identified the technical safety issues related to aging of major components and structures. The assessment results show that the conventional inservice inspection techniques and procedures sometimes have questionable capabilities in detecting age-related degradation and have inadequate capabilities in sizing aging damage. Emerging inspection and monitoring methods have been identified for detection, sizing, and trending of aging damage. Modifications of ASME Section XI codes and standards have been identified. The project has also identified several methods to mitigate aging damage in the major components and structures.

#### 4.5 DEVELOPMENT OF THE TECHNICAL BASIS FOR GUIDANCE TO SUPPORT REGULATORY ACTIONS INVOLVING NUCLEAR PLANT AGING

Objective: To develop technical guidance needed to implement the License Renewal Rule and to support future regulatory activities that address aging in nuclear power plants.

Contractor: Pacific Northwest Laboratory (PNL)

Methodology: In developing the technical basis for regulatory guidance and the supporting technical information, the approach has been 1) to collect and organize all available information related to age-related degradation in NPPs, and 2) to perform the integration and analysis required to apply this information to regulatory needs. The NPAR Program has been a major source of basic technical information for this topic. Additional sources include reports from other NRC research programs, industry technical reports, and a wide variety of reports that document results of domestic and foreign aging research programs. The collected body of information is summarized to a degree suitable for the required regulatory guidance. This information is then organized in an appropriate format to address the requirements in the proposed regulatory action, such as the LR Rule.

Status: Several products, some of which represent multiple revisions, have been generated. These products include 1) a detailed criteria and screening methodology for selecting SSCs important to LR; 2) criteria and requirements for programs for understanding and managing aging in support of LR; 3) the detailed technical basis for a regulatory guide on technical information requirements for LR applications; 4) a draft handbook that recommends practices to manage aging in NPPs; 5) organizing the current body of NPP aging technology in a format that corresponds directly with the Standard Review Plan (NUREG-0800); 6) a survey of the major regulatory instruments, including codes and standards, to assess adequacy with which they address age-related degra-



ation; and 7) a quantitative, deterministic ranking of safety-related PWR systems and their constituent components in terms of safety significance and aging effects on safety margins. Work continues on each of these activities.

Results: Guidance developed under this task has contributed to a draft regulatory guide (DG-1009) and NUREG/CR 5562, which supplements the regulatory guide and updated versions of regulatory instruments such as the Standard Review Plan. These documents represent the primary channels through which results of the NPAR program and other research on aging phenomena will be applied in the regulatory process. Anticipated products include the following:

- the technical basis for a regulatory guide that identifies the technical information to be supplied as part of an application for an operating LR; this regulatory guide will be needed to supplement the License Renewal Rule (55 FR 29043).
- a comprehensive handbook reviewing information useful for managing age-related degradation in NPPs; this handbook will supplement the regulatory guide with detailed, technical information presented in a systems format that mirrors the Standard Review Plan (NUREG-0800).

Plans: Development of the regulatory guide on technical content of LR applications will proceed in parallel with LR rulemaking. The goal is to issue a final regulatory guide in April 1992. The handbook will be complete in December 1991 and will be updated annually or biannually.

#### 4.6 TECHNICAL SPECIFICATIONS - FROM AGING PERSPECTIVE

Objective: To evaluate the lessons learned from NPAR aging studies that have potential application to technical specification (TS) requirements.

Contractor: Pacific Northwest Laboratory (PNL)

Methodology: The TS requirements will be evaluated from three perspectives: 1) their adequacy in considering age-related degradation mechanisms; 2) as potential contributors to aging (e.g., TS requirements for frequent fast starts of diesel generators of the loads imposed on auxiliary feedwater pumps while testing in the pumping mode), and 3) as a means for detecting, trending, and managing aging through surveillance testing. These evaluations will be performed by reviewing, from an aging perspective, the products of the NRC Technical Specification Improvement program and identifying potential modifications that would account for aging. This task will make recommendations for better testing and monitoring of the condition of SSCs and better detection of incipient failures. The emphasis will be placed on "quality over quantity" and frequency of testing.

Status: The revised TS have been subject to a recent comment period. Some NPAR studies are at a point where recommendations for TS improvement can be made. Other NPAR studies are in progress and identification of TS inputs is premature.

Results: The NPAR program provides for component- and system-specific identification of condition indicators and age-related degradation sites. The results of the TS assessment will be useful to focus on the quality of tests, such as detecting defects in the incipient or degraded state.

Plans: Key considerations are determining whether the TS effectively address detection and management of age-related degradation, and determining the extent that TS requirements contribute to age-related degradation as a result of test procedures.

#### 4.7 GUIDANCE FOR RECORDKEEPING

Objective: To determine if changes are needed in nuclear plant record-keeping practices to support aging management and potential extended plant operation. Such changes could affect the types of information collected, how it is stored, and how it is used.

Contractor: Pacific Northwest Laboratory (PNL)

Methodology: This task addressed the current status of nuclear plant records systems and the recordkeeping implications of NPAR research and similar programs in the United States and abroad. Also considered were the capabilities of recordkeeping techniques and technologies to make changes when needed. This research was conducted by reviewing the nuclear plant aging and records management literature and through discussions with nuclear plant records personnel.

Status: A draft NUREG/CR report<sup>(a)</sup> has been prepared for review; decisions concerning additional regulation needed in recordkeeping will be part of the overall LR rulemaking process. Although not formally representing the NPAR Program or the NRC, PNL staff participated in an IAEA Consultant's Meeting in July 1989, held for the purpose of drafting preliminary guidelines for recordkeeping needed to support nuclear plant life extension and LR.

Results: Plant records cannot resolve all the aging management issues. Although nuclear plant recordkeeping is extensive, implementation at different plants is highly variable and does not always fully support engineering, maintenance, and operational activities. Research on this task suggests that regulatory guidance and standards need to focus more on how recordkeeping can

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(a) Dukelow, J. S. 1990. "Recordkeeping Needs to Mitigate the Impact of Aging Degradation, Including During a License Renewal Period." (Draft), Pacific Northwest Laboratories, Richland, Washington.

support safe plant operation. Details about information that should be collected to support aging management and potential life extension require careful plant-specific analysis. Existing recordkeeping techniques and technology are adequate, but surveillance, monitoring, and testing techniques may not be adequate to determine residual life for all SSCs.

Plans: Future work on this task will include 1) continuing review of the recordkeeping implications of other NPAR aging assessments, 2) review of the treatment of recordkeeping needs in industry topical reports (prepared under the auspices of NUMARC), and 3) input for the technical basis for such regulatory guidance, if the NRC decides to issue formal regulatory guidance on recordkeeping needs for LR.

#### 4.8 THE USE OF NPAR RESULTS IN INSPECTION ACTIVITIES

Objective: To explore the areas in which NPAR Program results could enhance the NRC inspector's understanding of age-related degradation and contribute to a more effective inspection. Also, to identify the NPAR information that can focus inspections on those components and systems vulnerable to age-related degradation, and to develop a method for integrating research results into the inspection process.

Contractor: Brookhaven National Laboratory (BNL)

Methodology: Several tasks have been accomplished toward integrating NPAR information into the NRC Inspection Program. These tasks determined which NPAR information is relevant to the inspector, the way in which the NRC's Inspection Program addresses the performance of components and systems, and the format in which the NPAR information can be presented so that it can be readily accessed and updated.

Status: A report has been issued describing the NPAR results that can enhance NRC inspection activities. Recommendations are provided for communicating pertinent information to NRC inspectors. These recommendations are based on a detailed assessment of the NRC's Inspection Program and feedback from resident and regional inspectors. NPAR report summaries and inspection guides for aging have been completed for components and systems studied at BNL.

Plans: BNL will continue to work with the other NPAR contractors and with Office of Nuclear Reactor Regulation/Inspection and Licensing Program Branch to establish a complete manual of summaries and guides available to the NRC inspectors.

#### 4.9 AGING RELIABILITY MODELING OF ACTIVE COMPONENTS

Objective: To develop mathematical models using operating experience data to characterize the degradation rate of a component with its age; these models will estimate the component failure rate from this degradation rate and calculate the effects of the component aging on the plant risks.

Contractor: Brookhaven National Laboratory (BNL)

Methodology: The aging reliability modeling focuses on analyzing the times of component degradation to model how the degradation rates change with the age of the component. Similar trends are also developed for the failure rates as they change with the age of the component. The methodology also discusses the effectiveness of maintenance the applies to aging evaluations.

Status: The specific applications of this technique that are completed include residual heat removal pumps, service water system pumps, and compressors. Plant-specific data on a particular component are preferred when compared to that from the nuclear plant reliability data system. The study also discussed several limitations on the model and statistical tests for combining data for similar components from other plants.

Results: Models for other active components will continue to be developed. The failure rates developed will be used in the PRA models for specific plants to calculate risks caused by aging in these components. Finally, a screening methodology for selecting and prioritizing components will be developed to improve the present aging mitigation programs.

#### 4.10 REGULATORY INSTRUMENT REVIEW

Objective: To investigate how effectively regulatory instruments (documentation), such as regulatory guides, standards, specifications, and codes, provide guidance for aging of major safety-related LWR components.

Contractor: Pacific Northwest Laboratory (PNL)

Methodology: Eight regulatory instruments were chosen for review: the Code of Federal Regulations, the NRC Regulatory Guides, ASME Boiler and Pressure Vessel Code, generic safety issues, standard technical specifications, American Nuclear Society standards, the Standard Review Plan and Institute of Electrical and Electronic Engineers (IEEE) standards. Components for the review were selected from the NPAR program's major LWR safety-related component BWR and PWR plants. Typical components chosen for review were the reactor pressure vessel, primary piping, cables, emergency diesel generators, containment and basemat, and selected pumps and valves; these components are representative of mechanical, electrical, vessels/piping, and structural facilities. The review technique consists of determining the principal aging issues for each component, e.g., corrosion and fatigue, and then investigating the instruments to determine aging guidance with reference to the aging issue chosen. Results of the reviews have been reported in tables that list each component versus the instruments and aging issues.

Status: The review began in FY 1988. In March of 1989, a PNL report was published entitled "Guide to Regulatory Instruments for LWR Reactor Pressure Vessels: Aging and License Renewal Considerations" (Werry 1989). In FY 1990 a draft report, "Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components, Volume I" (Werry 1990), was issued for the following components: the reactor pressure vessel, the primary piping,

the steam generator, the pressurizer vessel, and the emergency diesel generator. During FY 1990, the instruments were evaluated for containment and basemat and cables.

**Results:** The results of the review have shown, with some exceptions found in the IEEE standards, that the instruments included in this review do not explicitly address aging or LR. However, aging management does exist because safety-related design, construction, and operation are consistent with principles needed to provide aging management and extended operation. The results indicate that revisions are needed in the instruments. Principal suggested revisions are functional criteria that clearly define the requirements of aging management. Other revisions are needed to enhance and encourage improvements in nondestructive examination (NDE) methods and tools.

**Plans:** The instrument review will continue through FY 1991 with the investigation of selected pumps and valves. The reviews of these components, together with the reviews of containment and basemat, will be published as Volume II of NUREG/CR-5490. Reviews of NRC notices and bulletins will be updated in FY 1991.

It is suggested that revisions to the regulatory instruments are needed to fully address the aging aspects of nuclear power plants. These instrument revisions will require additional information and research. More conclusive information is needed in the following areas: rate of time-dependent material degradation for SSCs, e.g., rate of consumption of fatigue design life; evaluation of aging initiatives in industrial/codes and standards for applicable revision; evaluation of replacement of SSCs as an aging management strategy. Research is also needed for improved NDE methods to evaluate material properties, including embrittlement, and fracture toughness, fatigue strength, and evaluation of intergranular stress corrosion cracking. (Improved acoustic emission<sup>(a)</sup> is a strong candidate for improved material assessment.)

#### 4.11 REVIEW OF INDUSTRY TECHNICAL REPORTS FOR LICENSE RENEWAL

**Objective:** Provide NRC with review comments on the eleven Industry Reports that identify LR issues for nuclear power plant components.

**Contractors:** All NPAR Laboratories.

**Methodology:** The reports are reviewed at the laboratories, and comments are compiled through the NPAR Program Coordinator for transmittal to NUMARC.

**Status:** See Figure 5.2 for the status of initial reviews. Further iterations on some of the NUMARC responses to NRC comments may be requested.

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(a) An ASME Boiler and Pressure Vessel Code, Section XI, Code Case N-471, is currently in the "pending adoption" use status for acoustic emission inservice inspection.

Results: The NPAR participants have provided comments on all Industry Reports (see Figure 5.2) to the NRC.

Plans: Further reviews will be conducted and additional comments provided to the NRC as requested.

#### 4.12 AGING EFFECTS IN BALANCE-OF-PLANT SYSTEMS

Objective: To identify and characterize aging effects in balance-of-plant (BOP) systems on plant performance, to calculate the plant risk due to systems aging, and to evaluate and assess the plant aging mitigation activities on BOP systems.

Contractor: Brookhaven National Laboratory (BNL)

Methodology: An aging assessment will be performed on the BOP systems and components that are considered important for safe operation of a plant. The study will analyze plant data pertaining to operating experience, unplanned reactor scrams, and maintenance to characterize aging in BOP components. The component data will then be used to determine the reliability of each BOP system.

Status: A preliminary evaluation of several other studies on BOP systems was completed. Unplanned reactor scrams and shutdowns reported in the NRC graybook (NUREG-0200) were obtained in database files for further analysis. The data pertaining to Combustion Engineering plants were analyzed. Similar efforts will be performed for the Westinghouse, Babcock and Wilcox, and General Electric plants to characterize aging in BOP systems.

Results: The frequency of unplanned reactor scrams has been cited as an indicator of safety performance. The main contributors to unplanned reactor scrams caused by BOP systems are the feedwater, main turbine, main generator, main steam (usually the steam bypass to the main condenser), and the condensate systems. The electrical distribution system and, less frequently, the circulating water, service/instrument air, fire protection, and heating, ventilation, and air conditioning (HVAC) systems are also frequent contributors. The electrical distribution system consists of the 120V ac power, switchyard, large plant loads, dc power and control centers. At a component level, the feedwater regulating valves and turbine-driven feedwater pumps are frequent contributors, as well as the main turbine electro-hydraulic control subsystem. Failures in the main generators are also important. Based on the study completed for Combustion Engineering plants, the reactor scram/shutdown rates initiated from support systems (such as the cooling water system; electrical distribution, other than the safety-bus; air systems; and HVAC, other than the control room) increase with the age of the plant. Similar trends are also observed for the turbine/generator systems.

Plans: In addition to analyzing the operating experience data, other considerations involving potential effects of failures of non-safety-related components will be analyzed. Plant-specific data relating to BOP systems will

be analyzed to calculate their aging effects on the plant risk. Plant maintenance activities will be assessed in the light of alleviating aging problems that can stem from the BOP system/component degradation.

#### 4.13 MAINTENANCE TO MANAGE AGING

Objectives: To 1) identify how aging management principles can be applied to improve maintenance practices; 2) interact with the ongoing efforts on risk-based prioritization to identify plant-specific applications of the risk-based methodology as it applies to maintenance practices; 3) systematically evaluate plant maintenance practices and determine how these practices need to be improved to effectively manage aging; 4) support NRC initiatives, including the Maintenance Rule, regulatory guides, and LR activities.

Contractor: Pacific Northwest Laboratories (PNL)

Methodology: The following four-fold approach is used. 1) Results of NPAR hardware tasks were reviewed to identify effective maintenance practices and recommendations for managing aging; detailed guidance is obtained from NPAR contractors concerning how results of their component or system studies should be translated into improved maintenance practices. 2) Surveys are made of maintenance practices in four areas--Japanese nuclear, U.S. commercial aviation, U.S. Air Force B-52 program, and the U.S. Navy Extended Operating Cycle Program for nuclear submarines. 3) NPAR prioritization methodology is used to identify maintenance intervals and other parameters that need to be evaluated on a plant-specific basis, including working with one or more plants to develop prototype maintenance program improvements, based on aging considerations. 4) Information from the NPAR component/maintenance system studies and from the plant-specific studies is used to identify specific program modifications (e.g., surveillance methods, intervals and trending) and to ensure that plant maintenance practices effectively address aging management.

Status: A draft report, Maintenance Practices to Manage Aging, A Review of Several Technologies, has been prepared, which includes summaries of maintenance practices in other technologies, including recommendations on how certain approaches may be useful to the nuclear industry. Preliminary discussions have been held between PNL maintenance and prioritization staff to identify important areas for future interaction.

Plans: Proceed to 1) compile and evaluate maintenance recommendations emerging from NPAR systems/components studies; 2) conduct selected evaluations of plant-specific maintenance data with relevance to prioritization methodology; 3) systematically review current plant maintenance methods and practices to identify areas where recommendations for changes are needed.

#### 4.14 AGING DEGRADATION OF CABLES

Objective: To determine the life-extension potential of cable products used in nuclear power plants and to determine the potential of condition monitoring for residual-life assessment of cables.

Contractor: Sandia National Laboratories (SNL)

Methodology: Twelve cable products, representative of those currently in use inside the primary containment of nuclear power plants, are being aged and then subjected to simulated design basis accidents. Simultaneous thermal and radiation aging exposure is used to age the cables to nominal equivalent lifetimes of 20, 40, and 60 years. This is followed by a sequential accident exposure consisting of high dose rate irradiation followed by a simulated loss-of-coolant accident steam exposure. Condition monitoring measurements performed during aging are assessed for their ability to predict cable degradation. Extensive monitoring during accident tests evaluates aged cable performance under accident conditions.

Status. Aging and accident tests of all cables have been completed. Extensive electrical and mechanical properties measurements have been and are being performed. A high temperature steam test, a submergence test, and a test of unaged cables have also been completed.

Results: The results of the Aging Degradation of Cables Program have shown that a number of popular cable products might qualify for extended life operation. However, some cable products failed during accident testing, while others exhibited reduced insulation resistance that could be significant in some applications. Some cable condition monitoring parameters could be useful to assess remaining cable life.

Plans: Additional condition monitoring measurements will be performed and the final reports for the experimental program will be completed.

#### 4.15 FIRE VULNERABILITY OF AGED ELECTRICAL COMPONENTS

Objective: To identify and investigate issues of plant aging that might result in an increase in plant fire risk.

Contractor: Sandia National Laboratories (SNL)

Methodology: The Fire Aging Program is drawing upon Sandia's unique fire testing and analysis capabilities developed, in part, as a result of the USNRC-sponsored Fire Protection Research Program, coupled with Sandia's extensive understanding of fire risk analysis methodologies and results. Using these capabilities, Sandia has identified and prioritized a list of six plant aging issues that can significantly affect estimates of fire risk. These issues are now being investigated in a systematic manner through testing and analysis.

Status: To date, two of the identified issues have been resolved. These are the impact of aging on 1) the vulnerability of electrical cables to thermal damage, and 2) the flammability of cable insulation materials. Current investigations of the impact of aging focus on the vulnerability of other types of plant equipment to fire-induced damage and the integrity and fire performance of passive fire barriers.



**Results:** The results of these efforts have demonstrated that two of the initially identified issues need not be further considered. The two studies involving the fire performance of electrical cables have demonstrated that changes in either the flammability or thermal fragility of cables that result from aging will not result in a significant increase in fire risk estimates.

**Plans:** Future efforts will continue to utilize Sandia's unique fire testing and analysis capabilities to investigate the remaining identified fire aging issues consistent with the prioritization of those issues that were developed early in the program.

#### 4.16 OTHER SPECIFIC TOPICS

Though not included in Table 4.1, the following have also been topics of special topics.

##### 4.16.1 Aging/Seismic Shock Interaction

An understanding of the vulnerability of degraded equipment to seismic disturbances is necessary for extending the design life of an NPP. This includes the original plant life of 40 years and any extended operating period. Current industry standards (IEEE Standards 323 and 344) and relevant regulatory guides require pre-aging before seismic qualification of electrical equipment. However, the NRC has not determined such a need for mechanical components and is currently evaluating the significance of aging as a factor in qualifying mechanical equipment. Therefore, an assessment is needed of the potential importance of aging in degrading the seismic performance of equipment.

Both the nuclear industry and the NRC have ongoing programs to assess the aging-seismic effects. These include the following:

- laboratory testing of naturally- and artificially-aged components
- qualifying equipment using existing test data
- using experimental data for qualifying components
- evaluating seismic fragilities for different components
- identifying weak links in certain equipment assemblies
- developing surveillance and maintenance programs to alleviate the effects of aging on seismic performance of equipment.

To avoid duplication of work, some studies involve coordination between industry and the NRC. Recently, it was determined that the qualified life of some equipment may have to be extended when a utility submits an application

to the NRC for a license renewal. In that case, the components originally qualified for a 40-year period would have to be reassessed for extended operation.

#### 4.16.2 Role of Safety-Centered Aging Management

Elements of maintenance work for NPAR aging studies on SSCs are outlined below:

- reviewing current practices and procedures used by nuclear utilities to maintain equipment.
- reviewing nuclear-equipment-vendor recommendations on maintenance of components or subcomponents selected for aging assessments.
- performing an evaluation, including a comparative analysis, of the relative merits of performing corrective maintenance when a component has been discovered to be malfunctioning versus the merits of programs concentrating on preventive maintenance. Emphasis is placed on the relationship between failures (causes or modes) expected to be experienced during operation and those that would potentially occur under the stresses associated with design basis or trigger events.
- evaluating the relative merits of predictive methods that can be used to identify imminent failures and to enable maintenance or replacement to be scheduled, based on actual equipment performance. This age-related approach lends itself to reliability methods and condition monitoring to mitigate equipment degradation.
- identifying, where possible, those component failure mechanisms likely to be inadvertently induced through faulty preventive or corrective maintenance actions. Specifically, identify those failures that might be detectable through short-term post-maintenance, surveillance, inspection, or monitoring.
- developing recommendations for acceptable or preferred maintenance practices, based on the preceding activities.

These activities emphasize the technical (e.g., mechanical, electrical) aspects of maintenance rather than institutional, organizational, programmatic, or human factor considerations.

Evaluations are made on the role of maintenance in managing the aging of SSCs. A review of present practices, in terms of a comparative analysis of corrective versus preventive maintenance, and recommendations for preferred practices are included for all SSCs in this study.

#### 4.16.3 Status of the Maintenance Rule

The importance of effective maintenance programs in ensuring safe and reliable operation of NPPs has been widely recognized by both the NRC and the industry. A high level of plant safety requires a high level of reliability of all safety-related SSCs. This high level of reliability must not only be designed into the SSCs and ensured by effective quality control procedures, it must be sustained during plant operation through sound maintenance practices (Dey 1989).

In the early 1980s, several industry and NRC studies concluded that there was significant room for improvement in planning and implementing effective maintenance programs. Shortcomings in the established practices at some NPPs were resulting in an unacceptably high rate of challenges to safety systems, and unnecessarily high radiological exposure of maintenance personnel. As a result of these findings, the NRC implemented a Maintenance and Surveillance Plan (SECY-85-129), beginning in 1985.

The in-depth findings and conclusions from Phase I of the NRC Maintenance and Surveillance Plan (MSP) confirmed that maintenance-related safety problems were present to varying degrees at different NPPs (NUREG-1212). In response to the findings of the MSP, the NRC developed and formalized the Commission's position on maintenance at NPPs (53 FR 9430). This policy statement was the subject of a public workshop (NUREG/CP-0099) and, together with findings from a review of U.S. and foreign maintenance programs (Dey 1988), formed the basis for issuance of a proposed rule (53 FR 47822) and a draft regulatory guide (DG-1001) defining the essential elements of an effective maintenance program.

Promulgation of the final Maintenance Rule was strongly opposed by the NPP industry, who had already initiated several actions aimed at improving the effectiveness of maintenance practices and increasing the reliability of NPPs. Consequently, the Commission decided to postpone rulemaking on maintenance for a period of 18 months (until June 1991) to monitor the industry initiatives and progress in improving the effectiveness of maintenance programs (54 FR 50611).

Activities to resolve the status of the Maintenance Rule have resumed. Options have been evaluated and a recommendation is scheduled to be presented to the ACRS in April 1991.

## 5.0 LICENSE RENEWAL

License renewal (LR) would permit qualified nuclear power plants (NPPs) to operate for a licensed period beyond the current 40-year limit. Since the mid-1980s, the nuclear industry has been actively developing the technical basis for LR. In parallel, the NRC has sponsored programs to understand the technical issues involved in LR and is developing the regulatory basis to review industry requests for LR. License renewal is a priority activity for the NRC and has been an element of the NPAR Program (see Figure 3.1 and Section 3.0).

Figure 5.1 shows the time frames for current licensed operation to 40 years and for proposed extended plant operation beyond 40 years. In the figure, aging processes are shown to be ongoing in both time frames and, hence, must be effectively understood and managed across the entire operating life of the plant.

### Aging

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Current Operating License

Extended Plant Operation

40

"Understanding Aging - A Key to Ensuring Safety"

"Managing Aging - A Necessity to Ensuring Safety"

Guy A. Arlotto

**FIGURE 5.1.** Nuclear Power Plant Licensing Regimes and Aging Importance

Although the NPAR aging studies have applied mainly to the operating license time frame, the principles of understanding and managing aging derived from the studies are equally applicable to extended plant operation. Specifically, the hardware studies (Section 3.0) and the special topics (Section 4.0) provide technical bases for understanding and managing aging in structures, systems, and components (SSCs). These studies are part of the technical bases that provide guidance and input to the NRC's development of the basis for LR. Some of the bases for understanding and managing aging are discussed in Appendix B.

## 5.1 REGULATORY GUIDANCE FOR LICENSE RENEWAL

The NRC has proceeded to establish the regulatory basis for LR by developing a LR Rule, which has been issued for public comment (55 FR 29043). The LR Rule will establish the regulatory guidelines for the LR process and will require that age-related degradation in NPPs be managed effectively during the LR period. The LR Rule will be supported by a draft regulatory guide to specify for the applicants the content and format for the LR application. This draft regulatory guide, Standard Format and Content of Technical Information for Nuclear Power Plant Operating Licenses, (DG-1009) also was issued for public comment (55 FR 50065). A third important initiative of the LR regulatory process will be the Standard Review Plan for License Renewal (SRP-LR), which is intended to guide the NRC staff in reviews of LR requests. The draft SRP-LR (NUREG-1299) also has been issued for public comment (55 FR 50065). To provide more detailed technical support to the regulatory guide and the SRP-LR, a compilation of aging management information (Morgan and Christensen 1991) has been prepared, including input from NPAR research staff at several laboratories. A draft (NUREG/CR-5562) has been issued for comment.

## 5.2 NPAR PARTICIPATION IN THE LICENSE RENEWAL PROCESS

Results from the NPAR aging assessments provide the regulatory staff with a basis to evaluate the effects of age-related degradation on safety-related SSCs both in current operations and in LR. The NPAR Program is contributing to the development of LR by providing technical support to the LR rulemaking and its implementation. Several NPAR initiatives support the License Renewal Rule:

- Lead responsibility for developing the draft regulatory guide on Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses (DG-1009)
- Lead responsibility for developing a Review of Information Useful for Managing Aging, NUREG/CR-5562 (Morgan and Christensen 1991, draft)

- Technical support to NRR
  - in preparation of the LR Rule and in resolution of public comments on the Rule
  - input to development of the SRP-LR
  - review of NUMARC industry reports (see Figure 5.2 for reports and review schedules)
  - potential input to prepare Safety Evaluation Reports (SERs)
  - potential support to review lead plant applications.

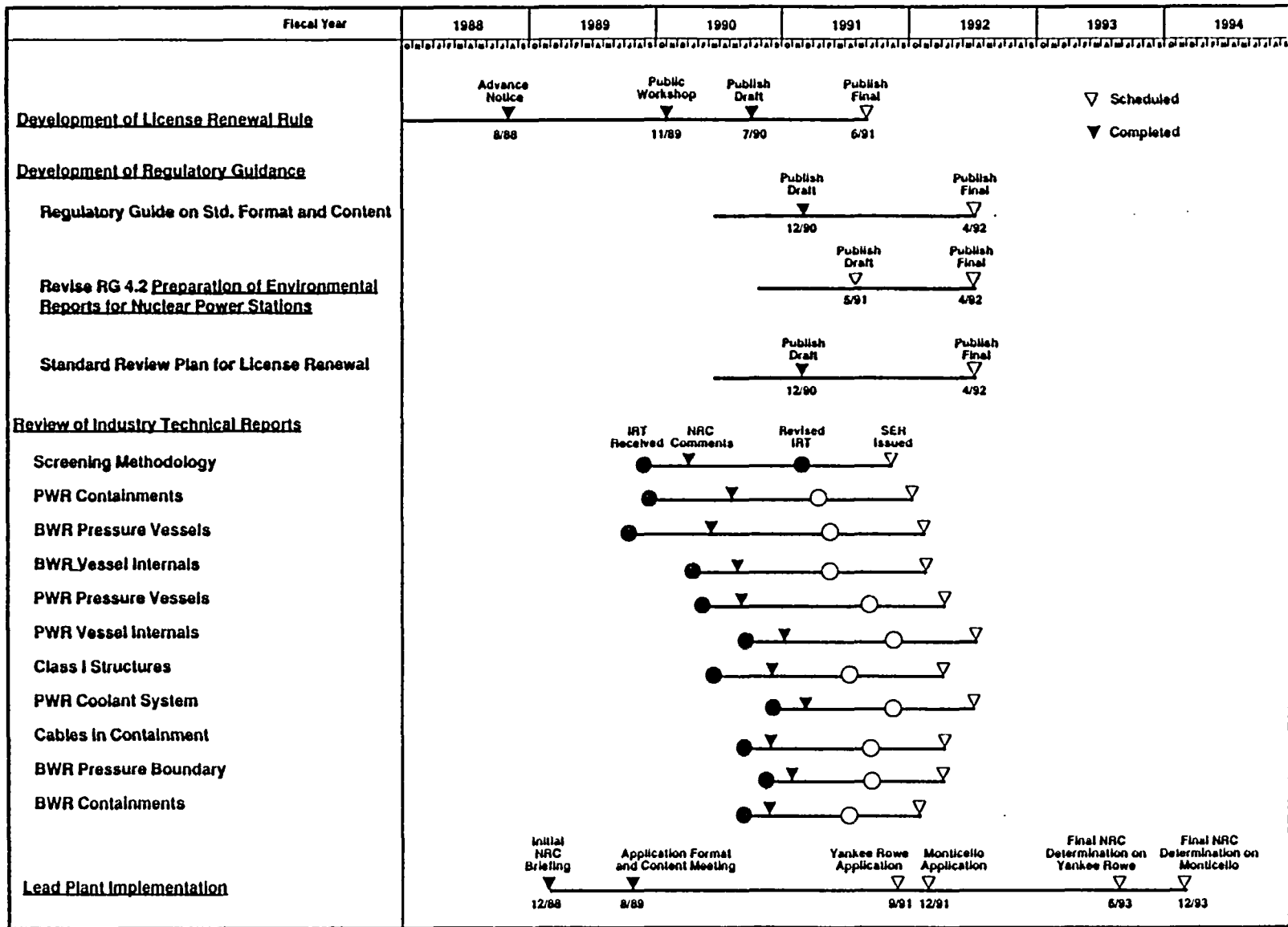
The following sections describe the industry and regulatory perspectives on LR, industry report reviews, lead plant implementation, and other NPAR roles.

### 5.3 INDUSTRY PERSPECTIVE

In the U.S. there are over 110 commercial NPPs in operation, constituting nearly 20% of U.S. electrical capacity. Several plants have operated for more than 20 years and a few are approaching 30 years. The nuclear industry is actively developing the basis to extend the life of operating reactors beyond their original license term. From the industry perspective, a major motivation for LR is economic. The projected net benefit to the U.S. economy is estimated to exceed \$100 billion, assuming a 20-year life extension for currently operating reactors. The industry recognizes that management of age-related degradation is a key consideration for LR.

License renewal initiatives have been developed through cooperation among the Department of Energy (DOE), the Electric Power Research Institute (EPRI), and utilities, coordinated by the working groups of the Nuclear Utility Management and Resources Council (NUMARC), Nuclear Utility Plant Life Extension (NUPLEX). Two pilot studies were completed: a boiling-water reactor (BWR) evaluation involving the Monticello plant (operated by Northern States Power Company) and a pressurized-water reactor (PWR) evaluation involving the Surry plant (operated by Virginia Electric Power Company). Subsequently, a bidding process was used to select two lead plants to develop formal applications for LR. The two plants selected were a BWR (Monticello operated by Northern States Power Co.) and a PWR (Yankee Rowe operated by Yankee Atomic Company). The LR applications are scheduled to be submitted to NRC in 1991 (see Figure 5.2).

From a technical perspective, it is recognized that certain plant SSCs important to LR that are not readily refurbishable or replaceable are subject to age-related degradation. The position was stated at the NRC Public Workshop on Technical and Policy Considerations for NPP License Renewal (NUREG-1411): "The focus of license renewal is mitigation and management of age-related degradation to ensure an adequate level of safety." It is



30102018.1

FIGURE 5.2. Schedule of License Renewal Rule Development and Implementation

commonly viewed that reliability and operational readiness of risk-significant SSCs subject to aging must be ensured during their entire operating life.

The industry, DOE, and EPRI have developed industry reports (IRs) that present, on a generic basis, the technical and safety issues that they consider important to LR. The reports have been subject to NRC review and comment. The industry proposes that preparations to renew licenses for individual plants would be guided by the IRs, but also that each plant would need to assess the plant-specific issues related to the components or structures addressed by the IRs. Both the industry and the NRC are focusing on programs to determine residual life of major LWR components and structures. A screening methodology to identify components that may degrade during extended operation has been proposed. The role of that screening methodology or an alternative approach will be an important LR consideration.

#### 5.4 REGULATORY RESEARCH PERSPECTIVE

Research programs conducted under the NRC Office of Research have focused on understanding and managing aging in nuclear safety-related SSC (see Figure 1.1). The Structural and Seismic Engineering Branch sponsors research on concrete structures; the Materials Engineering Branch is focusing on reactor pressure boundary components; the Electrical and Mechanical Engineering Branch conducts aging research on safety-related systems and components under the NPAR program. These programs help to resolve uncertainties regarding degradation processes that affect key SSCs and detection and management of degradation before safety is impaired. The programs also focus on methods to mitigate simultaneous, multiple age-related failures of components that could occur during a transient or accident, thereby compromising the function of the safety systems.

Reviews of operating experience indicate that components can fail from numerous causes, including corrosion, radiation, thermally-induced embrittlement of electrical insulation, pitting of electrical contacts, surface erosion, metal fatigue, oxidation, creep, and binding and wear. As operating reactors advance in age toward the nominal design life of 40 years or in extended operation, the major regulatory safety issues are ensuring that aged plant systems and components do not result in failures, including common mode failures that could weaken the defense-in-depth strategy, lead to an accident, or render inoperable the redundant but aged safety equipment needed to mitigate accidents.

##### 5.4.1 Technical Issues

The technical issues for a plant that demonstrates a satisfactory operating history over a 40-year period are expected to be related to the aging and degradation of hardware, not to the current licensing basis of the plant (NUREG-1362). Past and current regulatory programs, as well as the NRC's evaluation of abnormal operational events and inspection efforts, ensure that



design-related safety issues are resolved on a continuing basis, and modifications are made to the plant where necessary. These modifications, in both the physical plant and in the plant operating procedures, become part of the current licensing basis for the plant. The current licensing basis of the older operating plants is assumed to be adequate, although it may differ from the design basis of new plants.

Safety during the renewal term can be ensured by systematically assessing and managing the aging of selected safety-significant components of the plant. A systematic approach to assess and manage aging will ensure a uniformity in the scope of the licensees' efforts and in the standards that they should meet, while allowing the necessary degree of flexibility. The nuclear industry's pilot plant and lead plant studies emphasize the importance of aging analyses to support LR. Similar analyses will likely be conducted by all licensees applying for LR, and these analyses will take into account the significant findings of the aging studies by the NRC and the industry.

The aging management strategies for the renewal term adopted by the licensees will likely be plant specific, depending on plant type and vintage. However, the development and implementation of these strategies will depend on the quality of data and methods used in the aging analyses. Therefore, the NRC's specification of a formal methodology for these analyses will provide a high level of safety assurance during the LR term.

#### 5.4.2 Plans and Schedule

The schedule for implementing the proposed License Renewal Rule (Figure 5.2) includes separate but interdependent schedules for rule development, regulatory guidance development, IR reviews, SRP-LR development and lead plant implementation. The progress of each activity will influence the progress of the others. Appropriate participation by the industry has been assumed, based on schedules formally provided by NUMARC to the NRC.

Completion of rulemaking will provide the basis for review of the lead plants. If technical issues are resolved and regulatory guidance is developed in sufficient time to create a stable regulatory environment for LR, then other licensees can plan effectively for future generating capacity.

#### 5.5 INDUSTRY REPORT REVIEWS

While NRC developed the LR Rule and regulatory guidance, the nuclear industry (represented by NUMARC), EPRI, and DOE concurrently sponsored the development of several IRs on important aging mechanisms that can affect safety-significant SSCs. The NRC staff, assisted by the NPAR contractor laboratories, have actively reviewed the IRs as shown in Figure 5.2.

## 5.6 LEAD PLANT IMPLEMENTATION

Yankee Atomic Electric Company's Yankee Rowe plant (PWR) and Northern States Power Company's Monticello plant (BWR) are expected to be the first two plants to apply for LR. The preparation and application work at these plants is being sponsored jointly by DOE, EPRI, and NUMARC to test the LR process established by the NRC. A series of meetings involving the lead plants, the lead plant program sponsors, and the NRC has been initiated. Results from the NPAR aging assessments will contribute to the technical basis for evaluating the LR applications.

## 5.7 SUPPORT FROM CODES AND STANDARDS AND INDUSTRY GROUPS

Codes and standards organizations, particularly ASME and IEEE have been active in identifying methods to improve the codes and standards to develop guidelines that will be valuable to manage age-related degradation.

## 6.0 UTILIZATION OF RESEARCH RESULTS

The results from aging studies coordinated by the Nuclear Plant Aging Research (NPAR) Program have both technical and regulatory applications. This section summarizes the status of the results of these studies, the documentation, and the general and specific applications of results. Additionally, this section describes how the NPAR results contribute to understanding and managing aging and how they support license renewal (LR) rulemaking.

### 6.1 STATUS OF NUCLEAR PLANT AGING RESEARCH RESULTS

By the end of FY 1990, Phases I and II of the NPAR Program Strategy were completed for eight components and three systems. Additionally, Phase-I assessments of two components and one system were completed in FY 1990. One special topic investigation (Shippingport aging evaluation) was completed in FY 1990. Phase-I studies on fourteen special topics were completed in FY 1990. Results from these studies are contributing to the NPAR technical bases for LR. Two major NPAR initiatives supporting the LR Rule were completed in draft form in FY 1990.

### 6.2 STATUS OF NUCLEAR PLANT AGING RESEARCH DOCUMENTATION

Approximately 95 NUREG/CR reports have been issued (to April 1991). Brief summaries the NPAR reports that have been issued through May 1990 are provided in Kondic and Hill (1990). At that time, 72 NPAR reports had been issued and approximately 20 more were in draft form. Numerous papers based on NPAR results have been presented at the Water Reactor Safety Information Meetings, at technical society meetings, and at national and international symposia. The papers generally were published in peer-reviewed journals and in the proceedings of the respective conferences which include:

Third Workshop on Containment Integrity, Washington, D.C., May 21-23, 1986.  
NUREG/CP-0076, U.S. Nuclear Regulatory Commission, Washington, D.C.

Proceedings of an International Symposium on Safety Aspects of the Ageing and Maintenance of Nuclear Power Plants. 29 June to 3 July 1987, Vienna, Austria.  
International Atomic Energy Agency.

1988 ASME Pressure Vessel and Piping Conference, PVP 16. American Society of Mechanical Engineers, New York, New York.

Proceedings of the Topical Meeting on Nuclear Power Plant Life Extension,  
July 31 - August 3, 1988, Snowbird, Utah. American Nuclear Society, La Grange Park, Illinois.

Proceedings of the International Nuclear Power Plant Aging Symposium, ed. F. A. Beranek. NUREG/CP-0100. U.S. Nuclear Regulatory Commission, Washington, D.C.

Sixteenth Water Reactor Safety Information Meeting, ed. A. J. Weiss. NUREG-0097. U.S. Nuclear Regulatory Commission, Washington, D.C.

1989 ASME Pressure Vessel and Piping Conference, Honolulu, Hawaii, PVP Vol. 171. Am. Soc. Mech. Eng., New York, New York.

10th International Conference on Structural Mechanics in Reactor Technology (SMiRT), August 14-18, 1989, Anaheim, California. American Association for Structural Mechanics in Reactor Technology.

International Conference on Operability of Nuclear Systems in Normal and Adverse Conditions, September 18-23, 1989, Lyon, France. American Nuclear Society, La Grange Park, Illinois.

15th MPA-Seminar on Safety and Reliability of Plant Technology. Stuttgart, FRG, October 5-6, 1989. (To be published in Nucl. Eng. Design.)

Seventeenth Water Reactor Safety Information Meeting, ed. A. J. Weiss. NUREG/CP-0105, U.S. Nuclear Regulatory Commission, Washington, D.C.

Transactions of the Eighteenth Water Reactor Safety Information Meeting, ed. A. J. Weiss. NUREG/CP-0113, U.S. Nuclear Regulatory Commission, Washington, D.C.

The NPAR technical bases for LR and the overall research program have been subject to peer review, including reviewers from U.S. Nuclear Regulatory Commission (NRC) staff; the Advisory Committee on Reactor Safeguards (ACRS); the Nuclear Safety Reactor Research Committee; the NPAR laboratories; Equipment Quantification Advisory Group (EQAG) of the Electric Power Research Institute (EPRI); NSSS vendors; and other industry groups, codes and standards groups, and industry experts. Some NPAR results have been subject to review in workshops convened for detailed evaluations of the results.

### 6.3 TECHNICAL APPLICATIONS OF NPAR RESULTS

This section addresses the general and specific technical applications of the results from NPAR aging assessments.

#### 6.3.1 General Technical Applications

NPAR reports and other publications on aging assessments of safety-related systems, structures, and components (SSCs) represent technical bases that can assist both regulators and industry to understand further how aging affects equipment, how to improve management of the aging processes, and how to minimize potential impacts of aging on plant safety. Efforts are underway to improve accessibility of these technical bases.

Inquiries from industry and from NRC staff indicate that the NPAR technical findings are being utilized. The NPAR technical bases demonstrate a

sustained NRC commitment to understanding the effects of aging of SSCs, both for the current license period and for extended plant operation.

### 6.3.2 Specific Technical Applications

Results from the NPAR hardware-oriented engineering studies include several specific technical findings and recommendations that have near-term applications. The following are several examples:

- Identified deficiencies in current inservice testing (IST) program for auxiliary feedwater (AFW) pumps and published recommendations for improvements in the IST programs for AFW pumps [used for the development of American Society for Mechanical Engineers (ASME)-O&M Standards, Office of Nuclear Reactor Regulation (NRR) relief requests, and in the preparation of Bulletins 88-04 and 89-04].
- Published recommendations for preferred testing and maintenance of diesel generators, and generated technical data to support GSI-56, Diesel Reliability.
- Identified components (fuses, capacitors) that are prone to aging in battery chargers and inverters and need improved maintenance. Working to upgrade Institute of Electrical and Electronics Engineers (IEEE) Standard 650, Qualification of Class 1E Static Battery Chargers and Inverters.
- Published IEEE recommendations for Maintenance Good Practices for nuclear power plant (NPP) motors, motor operated valves (MOVs), solenoid operated valves (SOVs), and battery chargers/inverters; the recommendations contain significant uses of NPAR results.
- Provided direct support to NRR in resolution of service water system problems at a specific reactor.
- Provided major inputs from the NPAR Program to the proposed IEEE W.G.3.4 Guide on "Life Extension of Electrical and I&C Equipment."
- Published ASME/ANSI OM-16 draft on "Inservice Testing and Maintenance of Diesel Drives in Nuclear Power Plants," which contains key recommendations made in the NPAR-generated NUREG/CR-5078 (Lofgren et al. 1988).
- Transferred data developed in the NRC Equipment Qualification/Aging program to DOE for use in developing procedures for evaluating time-dependent dose-rate effects on four different cable materials.
- Transferred NPAR generated data to industry for developing snubber seal material recommendations.

- Identified degradation sites and mechanisms, stressors, and failure modes, and evaluated current inspection methods for twenty-two major light-water reactor (LWR) components: pressurized-water reactor (PWR) and boiling-water reactor (BWR) pressure vessels, PWR and BWR containments, PWR reactor coolant piping, safe ends, and nozzles, PWR steam generators, PWR reactor coolant pumps, PWR pressurizer, and surge and spray lines, PWR and BWR control rod drive mechanisms, PWR and BWR cables and connectors, PWR and BWR emergency diesel generators, PWR and BWR reactor vessel supports, PWR and BWR pressure vessel internals, PWR feedwater lines and nozzles, BWR recirculation piping and safe ends, BWR main steam lines and feedwater lines and nozzles, and BWR recirculation pumps. Results are being used to evaluate industry reports for license renewal considerations, preparing guidance related to understanding and managing aging, and preparing Standard Review Plans for License Renewal (SRP-LR).
- NPAR results related to fatigue failures in the field provided major input to the ASME Section XI report Metal Fatigue in Operating Nuclear Power Plants: A Review of Design and Monitoring Requirements, Field Failure Experience, and Recommendations for ASME Section XI Action. The report was prepared by the ASME Section XI Task Group on Fatigue in Operating Plants. One of the major recommendations, which was based on NPAR results, is to reevaluate the current inservice inspection (ISI) requirements and to include in the ISI program the inspection of highly stressed base-metal sites that are susceptible to fatigue damage.
- Developed a procedure using a linear aging model for the modification of probabilistic risk assessment to account for the effects of aging and used this method to carry out a risk-based prioritization of components and structures.
- Generated a single source document that delineates the regulatory documents governing the inspection and monitoring of reactor pressure vessels, steam generators, and major reactor piping, for use in both aging study and license renewal.
- Developed a novel diagnostic technique for MOVs based on the analysis of motor current signatures for remotely and nonintrusively detecting and differentiating among abnormalities, including those resulting from aging and improper maintenance.
- Developed a novel check-valve diagnostic technique based on the analysis of magnetic flux signatures for remote detection and trending of check-valve abnormalities, including those resulting from aging, improper operation, or improper maintenance. The technique has also been successfully demonstrated in industry tests conducted by the Nuclear Industry Check-Valve Users Group and is useful in satisfying check-valve IST requirements.

- Developed statistical models to analyze component degradation and failure data to understand the aging process of components. The modeling focuses on analyzing the times of component degradation to model how the rate of degradation changes with the age of the component. The modeling also considers the effectiveness of maintenance as it applies to aging evaluations.
- Contributed to formulation and implementation of Generic Letter 89-13 on SWS reliability.

#### 6.4 REGULATORY APPLICATIONS OF NPAR RESULTS

This section addresses general and specific regulatory applications of NPAR findings.

##### 6.4.1 General Regulatory Applications

The NPAR Program is providing technical bases for timely regulatory decisions regarding the effects of aging on NPPs of all ages. Informed oversight to ensure that aging of safety-related SSCs is effectively understood and managed is important to fulfilling NRC's regulatory responsibility. Through the NPAR utilization effort, NPAR results are communicated in a format that can assist the work of the NRC inspection staff.

##### 6.4.2 Specific Regulatory Applications

The NPAR Program is structured to respond to the following specific user-oriented needs:

- Develop data for identifying and resolving technical safety issues related to plant aging and LR.
- Support NRR and Office of Nuclear Regulatory Research (RES) in resolving generic safety issues involving aged plant safety systems, support systems, and electrical and mechanical components.
- Evaluate and recommend maintenance and surveillance methods needed to monitor aging of SSCs and to support LR.
- Provide information for developing inservice inspection procedures suitable for aged SSCs.
- Develop data and supporting information for revising appropriate NRC guidelines, technical specifications, and industry codes and standards.
- Provide technical information needed to prepare NRC Information Notices and Bulletins.

Revision to the body of regulations governing NPP operation is an ongoing process that applies key results from the NPAR aging research to improve the regulatory basis. Specific results are applied to the regulations and supporting documents in areas such as regulatory guides, technical specifications, generic safety issues (GSIs), and codes and standards. Specific cases are cited in the following sections.

#### 6.4.2.1 Overview of Specific Regulatory Applications

The support that NPAR results are providing to the regulatory process is summarized in Table 6.1. The table comprises a matrix indicating each of the ongoing and planned NPAR activities and the support (or potential support) for various user-oriented needs. The milestones and schedules for the various activities are presented in Section 9.0.

Specific examples of regulatory application of NPAR results are cited in the following sections.

#### 6.4.2.2 Generic Safety Issues and Regulatory Guides

One of the objectives of the NPAR Program is to support NRR/RES in resolving aging-related GSIs identified in NUREG-0933, "A Prioritization of General Safety Issues" (Emrit 1991). NUREG-0933 contains a recommended priority list to assist in the timely and efficient resolution of safety issues that have a high potential for reducing risk. Results from NPAR studies are contributing to resolution of several of these GSIs, including those listed previously in Table 3.3. For example, the NPAR Program has supported the resolution of the Generic Safety Issue B-56, "Diesel Reliability," by evaluating aging of emergency diesel generators and by publishing recommendations for preferred testing and maintenance of diesel generators. The NPAR results also supported the basis for resolving Generic Safety Issue A44 on Station Blackout. NPAR laboratory staff provided input to resolve Generic Safety Issue 51 on service water systems.

NPAR laboratory staff supported NRR to develop Generic Letter 89-13 on service water systems; also generic letters 89-04 and 89-10. The NRR has provided "users-need requests" to RES for resolving the GSIs. The NPAR Program is supporting NRR in resolving the Generic Safety Issue II.E.6.1, "Test Adequacy Study" and by assessing methods for monitoring motor-operated valves. The third issue in this category is GI-70, "PORV and Block Valve Reliability."

A residual life assessment is being performed as part of NPAR to evaluate the aging and residual life of major LWR components. The results of this task will indirectly support NRR in resolving several GSIs related to the primary reactor coolant system components. The results of this task will be used in resolving safety issues related to plant LR and in developing regulatory guidelines and review procedures for use by NRR in reviewing applications for LR.



**TABLE 6.1. Potential Use of NPAR Results, Involving Aging Consideration, for Systems, Structures, and Components**

<u>NPAP Research Activities</u>	<u>License Renewal</u>	<u>Generic Safety Issues</u>	<u>Maintenance and Surveillance</u>	<u>Plant Performance Indicators</u>	<u>Inspection</u>	<u>Codes and Standards</u>
Shippingport aging evaluation	X		X		X	
Risk evaluation of significant aging effects	X	X	X	X	X	
Residual life assessment	X		X		X	X
Containment structures	X	X	X		X	
Reactor vessel supports	X	X			X	
Snubbers	X	X	X		X	X
Control rod drive system	X		X	X	X	X
Piping systems	X	X	X		X	
High pressure ECCS	X		X	X	X	X
RHR/low pressure ECCS	X	X	X	X	X	X
Component cooling water system	X		X	X	X	X
Reactor coolant pumps (seals)	X	X	X		X	
Auxiliary feedwater system	X	X	X	X	X	X
Auxiliary feedwater pumps	X	X	X	X	X	
Service water system	X	X	X	X	X	
Heat Exchangers	X	X	X	X	X	
Electric motors	X		X	X	X	X
Motor-operated valves	X	X	X	X	X	X
Solenoid-operated valves	X		X		X	
Check valves	X	X	X	X	X	X
Power-operated relief valves	X	X	X		X	
Reactor protection system	X	X	X	X	X	X
Diesel generators	X	X	X	X	X	X
Class 1-E distribution system	X	X	X	X	X	X
Transformers	X	X	X		X	X
Chargers/inverters	X		X	X	X	X
Batteries	X		X		X	X
Circuit breakers	X	X	X	X	X	X
Bistables/switches	X		X		X	
Penetrations/connectors/cables	X	X	X		X	X
Compressors	X		X		X	
Bolts, attachment devices	X	X	X		X	

NPAR results contribute to updates to and development of regulatory guides. NPAR laboratory staff provided direct support to NRR in the revision of Regulatory Guide 1.9 on Diesel Generator Selection, Design, and Qualification and published a research information letter (RIL) on aging management in EDGs. Development of the draft regulatory guide (DG-1009), supporting the LR Rule, and NPAR support to development of the SRP-LR were discussed in Section 5.0.

#### 6.4.2.3 Maintenance and Surveillance

Effective maintenance and surveillance programs at NPPs are significant contributors to plant reliability. The NPAR Program supports the NRR Maintenance and Surveillance Program by evaluating the role of maintenance in managing aging effects, both generically under a special topic and specifically under the individual aging studies of SSCs.

Here are key recommendations from the NPAR aging assessments of SSCs that address maintenance:

- improve the approach to scheduling major maintenance for EDGs
- improve the flow conditions and types of measurements to be taken in conducting inservice testing of auxiliary feedwater and other safety-related pumps
- implement a comprehensive inspection and preventive maintenance program for inverters
- implement a preventive maintenance program which includes techniques for testing motors that yield trendable data
- conduct systematic inspection and maintenance for LWR metal containments.

The NPAR Program defines maintenance and surveillance needs to ensure the operational readiness of aged NPP safety systems and components and to support the NRR staff in reviews of requests for LR. The program also provides the aging-related information needed for developing maintenance program criteria and standards and maintenance indicators which can be used to monitor specific components and systems.

The NPAR Program also provided input to the NRC Maintenance Rule. The Maintenance Rule was discussed in a public workshop in November 1988. Subsequently, it was placed on hold for 18 months. Depending on its eventual status, the NPAR Program is prepared to provide further input, based on 1) insights to maintenance developed in NPAR SSCs studies, and 2) recognition that maintenance is important to effectively managing aging.

#### 6.4.2.4 Plant Inservice Inspection

NPAR results are applied to plant inservice inspections; for example, the signature analysis technique for MOVs is used to detect and differentiate among abnormalities, including those resulting from aging and from improper maintenance.

#### 6.4.2.5 Plant Inservice Testing

The NPAR research identified deficiencies in the current IST program for AFW pumps (Adams and Makay 1986; Kitch et al. 1988; Casada 1989); consequently, recommendations for improving the IST programs for AFW pumps have been published and are being used to develop ASME-O&M Standards; NRR Technical bases also were provided for relief requests and to prepare Bulletins 85-03, 88-04, 88-57 and 89-04. As a result of the electric motor NPAR research, it is recommended that certain motor parameters be monitored while performing required pump and valve inservice testing.

#### 6.4.2.6 NRC Inspections

An NPAR initiative is underway to support the following NRR programs: the Safety System Functional Inspection Program and the Generic Communication Program. In general, the Safety System Functional Inspection Program assesses whether plant modifications of selected safety systems have degraded the design margin to the point where the system's ability to mitigate design basis events is impaired. This program consists of an in-depth review of selected safety systems and is usually conducted at older plants. The objectives of the Generic Communication Program are:

- to inform licensees of problems, including those caused by aging, that have developed in individual plants
- to require action when these problems are shown to be significant and generic.

These programs apply to the pressure boundary hardware, drivers, actuators, electrical power, and the instrumentation and controls of engineering safety features (ESFs).

The NPAR Program supports NRR in developing inspection procedures that are relevant to aging; NRR includes these procedures in the Inspection Enforcement Manual issued to guide the activities of the regions. For example, some inspection procedures establish guidance for ascertaining whether inservice inspection and testing activities are programmed, planned, conducted, recorded, and reported in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

The NPAR Program is developing initiatives to support ongoing inspections conducted by the regional offices under the NRR inspection program. These inspections ensure that systems and components have not degraded beyond acceptable margins from any cause, including aging. To provide the inspection

staff with the current status of NPAR research results, the results from the NPAR program will be summarized and provided to the regional and NRR inspection staff.

#### 6.4.2.7 Codes and Standards

Codes and standards groups provide important input to define the technical requirements for selected NPP electrical and mechanical components to ensure their operational integrity. Through participation in technical committees, NPAR participants contributed to developing recommendations to revise relevant ASME and IEEE codes and standards. These codes and standards address the safe operation of NPPs as component and system service times increase. Examples of the application of NPAR Program results in the improvement of codes and standards include the following:

- The proposed ASME Section XI recommendation for inspection of highly stressed sites in base metals susceptible to thermal fatigue damage.
- IEEE recommendations for Good Maintenance Practices for NPP electrical equipment on motors, MOVs, battery chargers/inverters, and SOVs contain significant use of NPAR results.
- The proposed IEEE W.G.3.4 Guide on "Life Extension of Electrical and I&C Equipment" will contain major inputs from NPAR Program participants.
- ASME/ANSI OM-16 draft on "Inservice Testing and Maintenance of Diesel Drives in Nuclear Power Plants" contains key recommendations made in the NPAR-generated NUREG/CR-5078, Vol. 2 (Lofgren et al. 1988).
- NPAR results are being used to upgrade IEEE Standard 650, "Qualification of Class 1E Static Battery Chargers and Inverters."
- IEEE Working Groups 4.2 is revising IEEE Standard 387 to include requirements of IEEE Standard 749 so that criteria and testing requirements for emergency diesel generators will be in the new document. NPAR data and information is an important resource for this work.
- ASME/ANSI OM-4 "Examination and Performance of Nuclear Power Plant Dynamic Restraints (Snubbers)," provides general and service-life monitoring guidelines to the OM-4 standard.
- ASME/ANSI OM-6 "Inservice Testing of Pumps in Light-Water Reactor Power Plants" includes recommendations regarding recordkeeping, data collection, and acceptance criteria based on NPAR research into auxiliary feedwater pumps (Adams and McKay 1986; Kitch et al. 1988).

- ASME/ANSI OM-8 "Startup and Periodic Performance Testing of Electric Motor Operators on Valve Assemblies in Nuclear Power Plants" includes recommendations regarding recordkeeping, data collection, and acceptance criteria based on NPAR research into MOVs (Greenstreet, Murphy, and Eissenberg 1985; Haynes 1989).
- ASME-ANSI OM-10 "Inservice Testing of Valves in Light Water Reactor Power Plants" includes recommendations regarding recordkeeping and stroke time testing criteria and methodology based on NPAR research into MOVs (Greenstreet, Murphy, and Eissenberg 1985; Haynes 1989).
- ASME OM Administrative Standard 6200, "Format and Content of Standard Parts," and Administrative Standard 9100 "Glossary of Terms" were prepared with major input from the NPAR program.
- ASME CS 860900 "A Guide to Writing ASME Codes and Standards" was prepared with major input from the NPAR program.

The Special Working Group on Life Extension, ASME Section XI, is coordinating activities related to initiatives of interest to the NPAR Program. A special IEEE working group was established to prepare a guide for life assessment of Class 1E electrical controls. The components and systems currently of interest and being considered in ASME standards are listed in Table 6.2. Some of the relevant IEEE standards are listed in Table 6.3. Periodic briefings and information exchanges with appropriate codes and standards committees are scheduled as part of NPAR.

## 6.5 APPLICATION OF NPAR RESULTS TO UNDERSTAND AND MANAGE AGING

The body of NPAR data that is available to understand and manage aging draws from both aging investigations of SSCs, using the NPAR strategy (Section 3.0 and Figure 3.1), and from the special topics (Section 4.0 and Table 4.1).

### 6.5.1 Results from Aging Investigations on Components and Systems

Results from the Phase-II comprehensive aging assessment on the components and systems are illustrated in this section in briefs that address how the work was carried out and how the results apply to understanding and managing aging.

#### 6.5.1.1 Understanding and Managing Aging in Emergency Diesel Generators

Understanding aging in EDGs is complex because the equipment is large and comprises many subsystems and components. Also, nine EDG suppliers are represented in U.S. NPPs. A laboratory approach to conduct the aging investigation was not practical, so an alternative approach, summarized below, was devised.

**TABLE 6.2.** Selected ASME Standards for Operation and Maintenance of Mechanical Equipment

<u>Standard Number</u>	<u>Title</u>
ASME OM-1	Inservice performance testing of nuclear power plant pressure relief devices.
ASME OM-2	Requirements for performance testing of nuclear power plant closed cooling water systems.
ASME OM-3	Requirements for vibration testing of nuclear piping systems.
ASME OM-4	Examination and performance of nuclear power plant dynamic restraints (snubbers).
ASME OM-5	Inservice monitoring of core support barrel axial preloads in PWRs.
ASME OM-6	Requirements for performance testing of pumps in light-water cooled nuclear power plants.
ASME OM-7	Thermal expansion testing of nuclear piping systems.
ASME OM-8	Requirements for preoperational and periodic performance testing of motor-operated valve assemblies.
ASME OM-10	Requirements for inservice testing of valves in light-water cooled nuclear power plants.
ASME OM-13	Requirements for periodic testing and monitoring of power-operated relief valve assemblies.
ASME OM-14	Requirements for vibration monitoring of rotating equipment.
ASME OM-15	Requirements for performance testing of nuclear power plant emergency core cooling systems.
ASME OM-16	Inservice performance testing of nuclear power plant diesel drives.
ASME OM-19	Startup and periodic testing of electro-pneumatic-operated valve assemblies used in nuclear power plants.

Approach to Understanding EDG Aging - A panel of Pacific Northwest Laboratory (PNL) and industry EDG experts analyzed approximately 2000 EDG failure cases from nuclear databases according to failure cause and relation to aging. The following actions were taken:

**TABLE 6.3. IEEE Standards for Electrical Equipment for Nuclear Power Plants that Provide Information Relevant to Aging and LR**

Standard Number	Title
IEEE 308	Criteria for Class 1E power systems for nuclear power generating stations.
IEEE 317	Electrical penetration assemblies in containment structures for nuclear power generating stations.
IEEE 323	Qualifying Class 1E equipment for nuclear power generating stations.
IEEE 334	Standard for type test of continuous duty Class 1E motors for nuclear power generating stations.
IEEE 336	Installation, inspection, and testing requirements for Class 1E instrumentation and equipment at nuclear power generating stations.
IEEE 338	Standard Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems (ANSI)
IEEE 344	Recommended practices for seismic qualification of Class 1E equipment for nuclear power generating stations.
IEEE 382	Standard for qualification of safety-related valve actuations.
IEEE 383	Standard types tests of Class 1E electric cables, field splices, and connections for nuclear power generating station.
IEEE 387	Criteria for diesel generator units applied as standby power supply for nuclear power generating stations.
IEEE 501	Seismic testing of relays for nuclear power generating stations.
IEEE 535	Qualification of Class 1E lead storage batteries for nuclear power generating stations.
IEEE 572	Qualification of Class 1E connection assemblies for nuclear power generating stations.
IEEE 549	Qualify Class 1E motor control centers for nuclear power generating stations.
IEEE 649	Qualification of 1E motor control centers
IEEE 650	Qualification of Class 1E static battery chargers and inverters for nuclear power generating stations.
IEEE 749	Periodic testing of diesel-generator units a applied as standby power supplies in nuclear power generating stations.
IEEE 944	Application and testing of uninterruptible power supplies for nuclear power generating stations.
IEEE/ANSI C37.13-1981	Low voltage ac power circuit breakers used in enclosures.

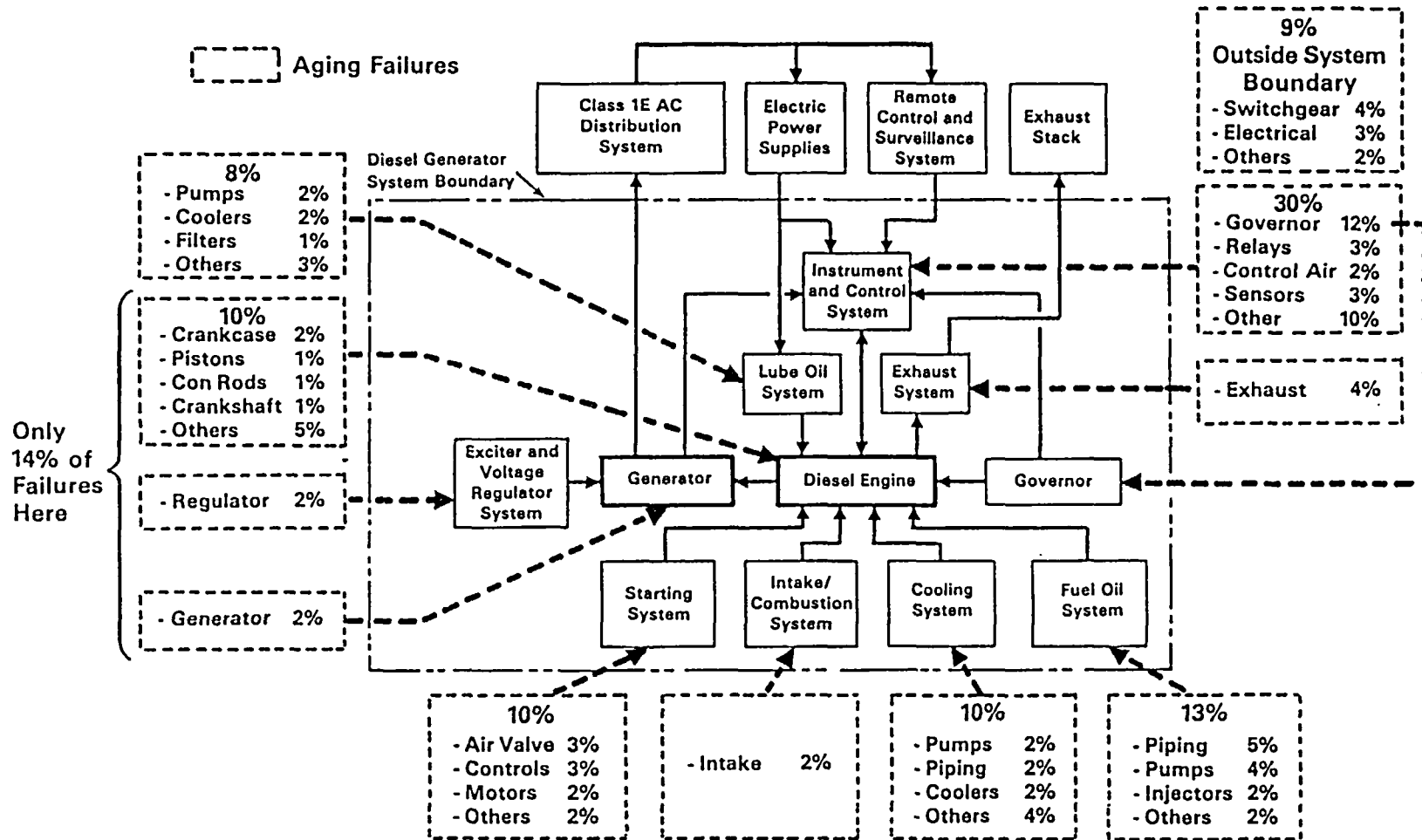


FIGURE 6.1. Emergency Diesel Generator Subsystem and Component Failure Rates



- Development and Application of a Computerized Database - a database was developed, sorted according to engine type, failure cause, and aging susceptibility of subsystems and components.
- Validation of Results - the results were validated in a PNL/industry workshop and by extensive peer review.
- Documentation of EDG Aging Assessment - the EDG aging assessment was documented in NUREG/CR-4590 (Hoopingartner et al. 1987; Hoopingartner and Vause 1987).
- EDG Aging Characteristics - Figure 6.1 identifies the EDG subsystems and components that are most susceptible to aging.
- Conclusions - Key conclusions include 1) the principal aging mechanisms<sup>(a)</sup> in nuclear EDGs are vibration, vibration loosening, thermal and physical shock; 2) the EDG systems most vulnerable to aging are instrument and control, fuel, starting system, cooling system, and 3) fast-start testing contributes to EDG aging (also acknowledged earlier in NRC Generic Letter 83-41). PNL participation in an EDG engine examination provided evidence of advanced engine wear due to fast starts.

The following are key outcomes from the EDG aging research:

- development of recommendations to mitigate EDG aging in a PNL/EDG consultant workshop
- documentation of the recommendations and extensive peer review
- further validation of the results by in-depth discussions during plant visits to three utilities
- publication of the proposed recommendations in NUREG/CR-5057 (Hoopingartner and Zaloudek 1989)
- recommendations to 1) replace the fast-start statistical basis for EDG reliability assessment with an engine "health check" approach, 2) permit slower starts and longer runtimes and monitor and trend several parameters that indicate engine condition, and 3) conduct major engine overhauls based on need rather than elapsed time.

#### 6.5.1.2 Understanding and Managing Aging in Electric Motors

Despite the fact that modern electric motors are better built and more sophisticated than older electric motors and maintenance practices are more extensive, electric motors still fail. Most motor failures described in the

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(a) Varies with engine type, subsystem, and utility.

operating experience data bases for U.S. NPPs occurred during starting. For smaller motors (under 200 hp), which constitute more than 90 percent of the motor population in a typical NPP, the stator insulating system and bearing assemblies were the subcomponents that failed most frequently. This accounts for almost 70 percent of the reported failures. Large motors, on the other hand, are equipped with devices to monitor the motor insulation and bearing and oil temperatures. Corrective actions are taken immediately on indication of excessive insulation temperature or rising bearing or oil temperature. Therefore, insulation and bearing failures are not as pronounced in large motors. The major factors contributing to large-motor failures include voltage surges and mechanical stresses caused by centrifugal or magnetic forces.

Understanding Motor Aging. A review of motor testing techniques documented in IEEE standards and related literature revealed a number of test methods available to monitor the condition of stator winding insulation and bearing assemblies. The study has also identified the predominant failure modes and mechanisms caused by aging and service wear effects. The insulating material is degraded primarily by heating cycles of the winding because of starting and overload conditions. A humid environment will produce more rapid degradation in both cases. The bearing failures are primarily caused by lubrication deterioration or bearing misalignment. Using cost-effective techniques to monitor these two problems could eliminate many untimely failures and, thus, improve the overall system reliability. The following information is useful for understanding electric motor aging:

- Documentation of Motor Aging Assessment - NUREG/CR-4156, (Subudhi, Burns, and Taylor 1985).
- Motor Aging Characteristics - Figures 6.2, 6.3, and 6.4 identify the motor subcomponents that are susceptible to aging.

Managing Motor Aging. To determine the tests or analysis methods that can aid in identifying age-related material degradation in motor components at an incipient stage, two test programs were conducted. One involved periodic testing of a 10-hp, 460-V induction motor with 12 years of service in a commercial nuclear plant. The motor was subjected to plug-reverse cycling to induce age-related degradation and monitored for various test parameter trends with age (number of reverse cycles). The second test program included diagnostic testing of a failed 400-hp motor with a service life of more than 20 years in a nuclear research reactor. The form-wound coils of the stator windings were tested to determine their condition by comparing similar test parameter variations. Based on these two test programs and industry operating experience trends, insulation and bearing tests are recommended as functional indicators. These indicators, when trended with the age of the motor under surveillance, would indicate the rate of deterioration occurring as a result of aging and service wear. Information on managing motor aging includes:

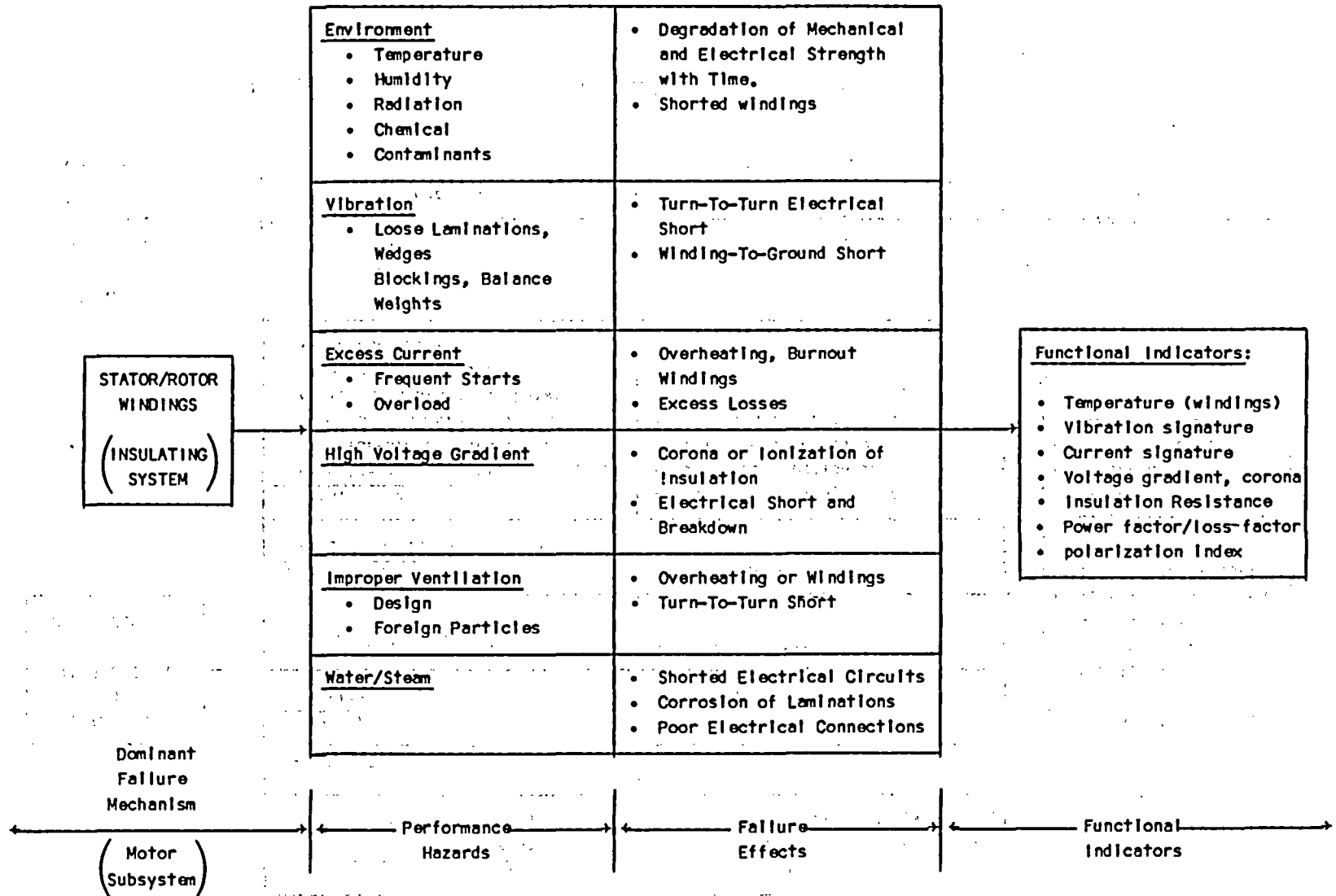


FIGURE 6.2. Functional Indicators for Dielectric Integrity

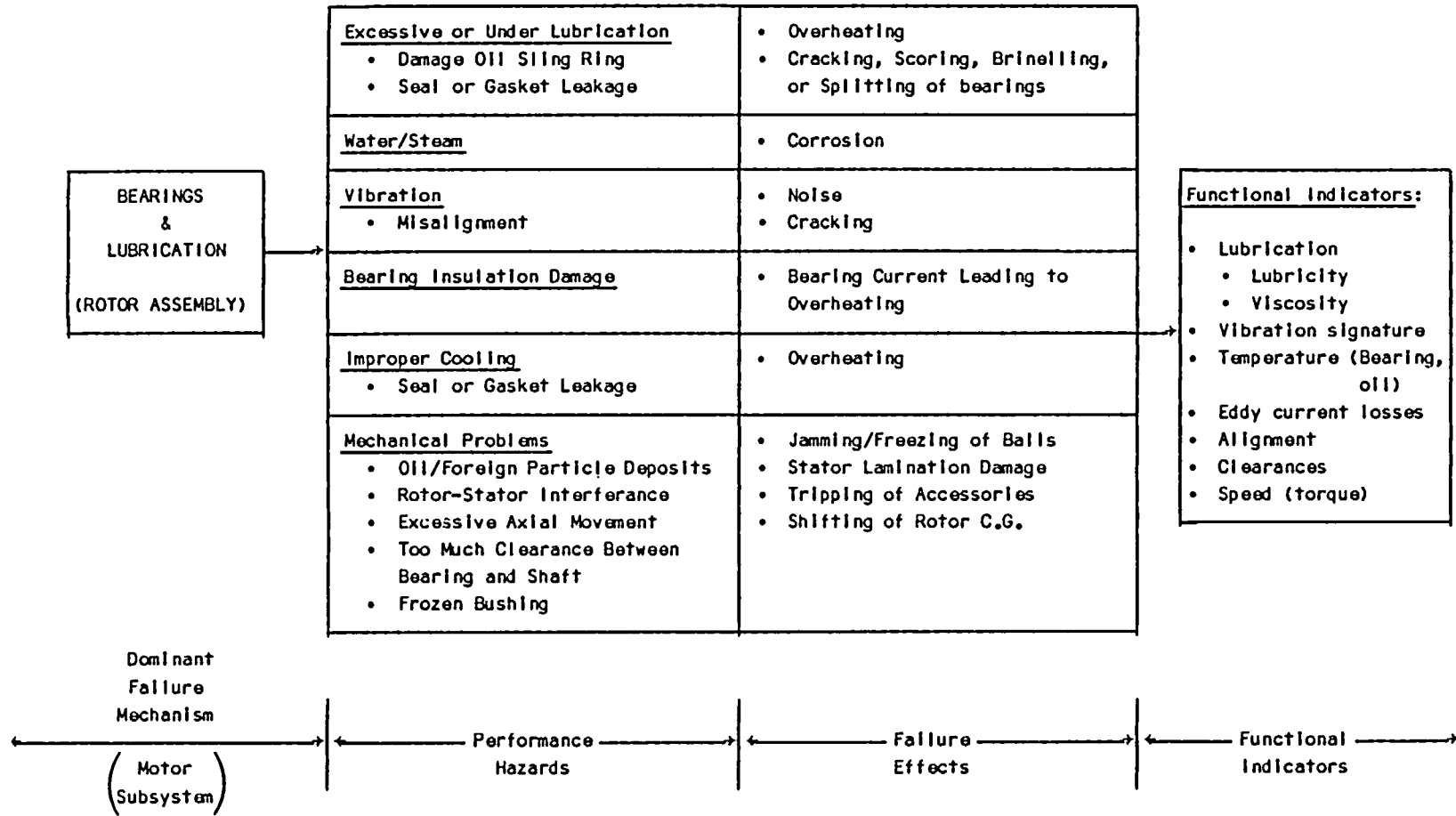


FIGURE 6.3. Functional Indicators for Rotational Integrity

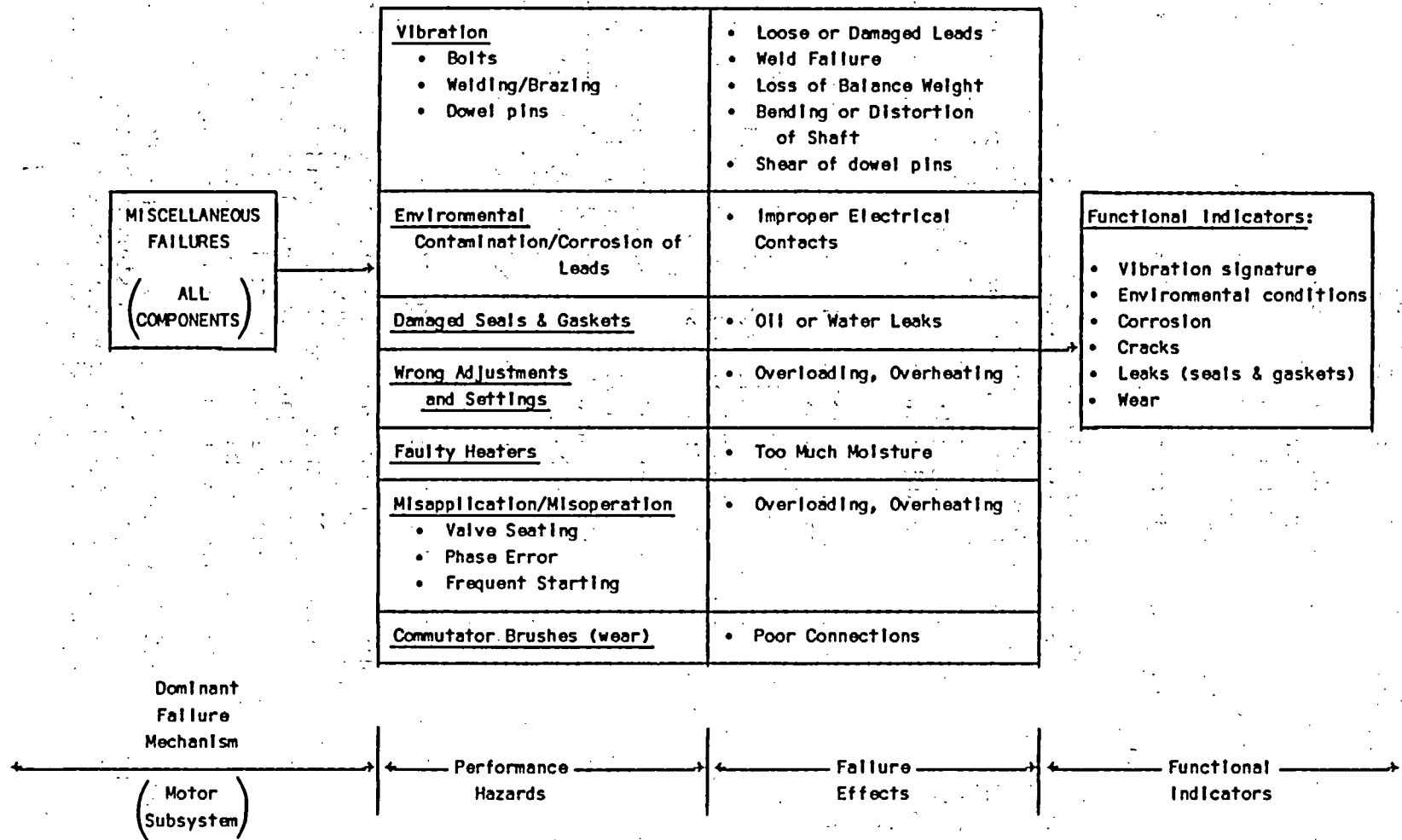


FIGURE 6.4. Functional Indicators for Mechanical Integrity

- Documentation of Motor Aging Management - NUREG/CR-4939 (Subudhi et al. 1987a, 1987b; Subudhi, Taylor, and Sheets 1987)
- Methods of Aging Management - Monitoring age-related degradation in electric motors can be achieved by instituting an effective maintenance program that includes periodic testing of the subcomponents. The two test programs discussed provide useful results on various dielectric and bearing tests.

#### 6.5.1.3 Understanding and Managing Aging in Battery Chargers and Inverters

The dominant causes of failure associated with nuclear power plant battery chargers and inverters are overheating, electrical transfers, and personnel errors. Overheating can degrade the life of components and materials. In many cases, the stresses resulting from these occurrences result in an accelerated aging of critical components. For example, electrolytic capacitors that provide electrical filtering were found to be temperature sensitive. Their failure in a short circuit mode results in the direct loss of the equipment's availability.

To manage the effects of aging in battery chargers and inverters, Gunther, Lewis, and Subudhi (1988) presented a comprehensive program based on an evaluation of the effectiveness of current storage, maintenance, repair, and replacement practices. This program addresses the integrity of discrete components such as capacitors, transformers, and semi-conductors and other entities such as cable connectors, wiring, and structural fasteners.

While emphasis has been placed on maintenance practices for managing plant aging, the potential improvement in vital bus reliability through the use of an automatic transfer switch should be considered. Other recommended design improvements discussed by Gunther, Lewis, and Subudhi (1988) include the use of equipment for detecting and suppressing electrical bus disturbances, the use of higher voltage and temperature-related components, and the addition of forced air cooling.

#### 6.5.1.4 Understanding and Managing Aging in Batteries

The most common aging problem for lead-acid storage batteries is thermally induced oxidation of the grids and top conductors that are made of a lead alloy. Lead experiences a 21 percent growth as it oxidizes to lead dioxide. This growth causes the plates (including grids) to swell, causing poor contact between the grid and the active material in the plate, and results in decreased capacity of the battery. In addition, oxidation causes the lead to become brittle, which can lead to decreased seismic ruggedness.

Edson and Hardin (1987) and Edson (1990) reported the results of a comprehensive aging assessment program for lead-acid storage batteries. The program included seismic testing of naturally-aged batteries that were about 13 years old to evaluate the effectiveness of current surveillance, monitoring, and replacement practices. From seismic testing, it was concluded that

when batteries are maintained and operated in accordance with IEEE Standard 450, Regulatory Guide 1.129, and manufacturer's recommendations, little, if any, electrical capacity will be lost as a result of seismic shaking at levels that are typical of the most severe seismic event postulated for NPP sites in the United States.

### 6.5.2 Results from Nuclear Plant Aging Research Special Topics

Several NPAR special topics from Table 3.1 have made significant findings contributing to understanding and managing aging, illustrated in the following sections.

#### 6.5.2.1 Effects of Aging on Risk

Results from NPAR aging/risk studies are utilized in the development of a procedure, based on the linear aging model, for modifying probabilistic risk assessments (PRAs) to account for the effects of aging on reactor safety. This procedure has been used to conduct risk-based prioritization of SSCs and is being used in PRA calculations to quantify the effects of aging.

Key findings from the aging/risk studies are summarized here:

- NUREG/CR-5248, (Levy et al. 1988) "Prioritization of TIRGALEX-Recommended Components for Further Aging Research," was issued in November 1988. Thirty safety-related plant components and structure groups were prioritized by an expert panel using the Risk Significance of Component Aging and Aging Management Practices (RSCAAMP) methodology, providing insights to the relative safety significance of the equipment that was addressed.
- The results of the RSCAAMP prioritization workshop showed that aging effects (e.g., aging failure rate and the weaknesses in maintenance programs to mitigate aging) can be major risk contributors.
- The RSCAAMP methodology was enhanced by incorporating the interactions of multiple components that had been subject to aging.
- This methodology, which accounts for multiple component aging effects, was incorporated into software that enabled analyses of the risk contributions due to aging of components in the minimal cut sets of probabilistic risk assessments.
- The aging effects software were applied to PRAs for a PWR and a BWR to generate a parametric study that assessed the impact of aging on total core melt frequency (C). Aging resulted in significant increases (factors of 10 to 1000) over the baseline plant core melt frequency ( $\Delta C$ ) for maintenance programs reflected by component renewal periods of from 18 months to 72 months and for an aging failure rate corresponding to one failure after 5 years. The interaction of multiple aged components was a major cause of the increased  $\Delta C$  at the renewal interval of 72 months.

### 6.5.2.2 Residual Life

Residual life assessments for major LWR components include preparing summary charts that highlight information useful for understanding and managing aging. The summary charts identify degradation sites, aging stressors, mechanisms, and failure modes; requirements and recommendations pertaining to inservice inspection, surveillance, and monitoring; and actions useful for the mitigation of aging of major LWR components. Figures 6.5 through 6.15 show summary charts for the following components:

- Emergency Diesel Generators (Figure 6.5)
- PWR and BWR Pressure Vessels (Figures 6.6 and 6.7)
- BWR Mark I containments (Figure 6.8)
- PWR and BWR Pressure Vessel Internals (Figure 6.9 and 6.10)
- PWR Cooling System Piping and Nozzles (Figure 6.11)
- PWR Steam Generator Tubes (Figure 6.12)
- Pressurizer, Surge, and Spray lines (Figure 6.13)
- BWR Recirculation Piping (Figure 6.14)
- LWR coolant pumps (Figure 6.15).

### 6.5.2.3 Shippingport Aging Assessment Results

Decommissioning of the Shippingport PWR afforded a unique opportunity to conduct onsite assessments, acquire a wide range of naturally-aged components, and obtain a variety of radiological and materials specimens for testing. The scope of the Shippingport aging assessment is outlined below:

- in-situ measurements of circuit integrity on 42 electrical circuits
- in-situ measurements of ferrite contents of cast stainless steels to select specimens for subsequent studies of thermal embrittlement
- on-site drill operations to obtain eleven 15 cm-diameter specimens from the RPV neutron shield tank to evaluate low-dose neutron embrittlement of carbon steel
- acquisition and shipment to participating laboratories of approximately 200 naturally-aged components, associated manuals, operating histories, etc.
- examples of technical and regulatory results that are emerging from the Shippingport investigations are:



# Understanding and managing aging of emergency diesel generators

## Principal Diesel Engines in Nuclear Service

### Manufacturer

- ALCO
- Allis Chalmers
- Caterpillar
- Cooper Bessemer
- Fairbanks Morse
- Electro-Motive Division
- Nordberg
- Transamerica DeLeval
- Worthington
- Others

### Materials: (typical)

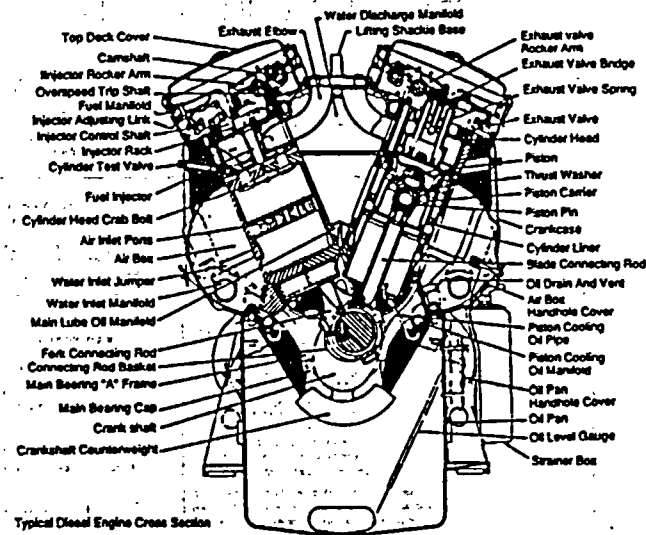
Alloy steels, welded steel plates, castiron including gray, aluminum, stellite seats, forged steels, ductive irons, non metallic gaskets, hoses, seals

### Capacities:

HP 215 to 670 per cylinder or 800 HP to 8390 HP  
KW ratings 50, 500, 1200, 3000, 6000

### Stressors:

Cooling water, lubricating oil, fuel oil, starting air, intake and exhaust, deterioration, dynamic stress, vibration, thermal fatigue, wear and harsh testing

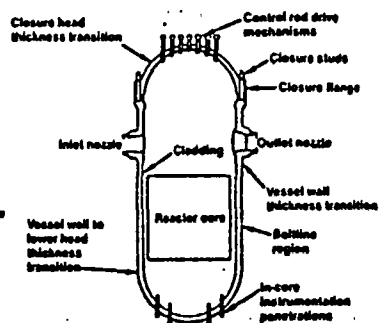


UNDERSTANDING AGING (Stressor & Environment Interactions)		MANAGING AGING	
Sites	Aging Concerns	Inservice Inspection, Surveillance	Mitigation
<ul style="list-style-type: none"> <li>• Instrument and control systems                             <ul style="list-style-type: none"> <li>Governor</li> <li>Sensors</li> <li>Relays</li> </ul> </li> <li>• Startup component                             <ul style="list-style-type: none"> <li>Piping on engine</li> <li>Injector pumps</li> </ul> </li> <li>• Starting system                             <ul style="list-style-type: none"> <li>Controls</li> <li>Starting air valve</li> <li>Starting motors</li> <li>Air compressor</li> </ul> </li> <li>• Switchgear system                             <ul style="list-style-type: none"> <li>Breakers</li> <li>Relays</li> <li>Instrument and controls</li> </ul> </li> <li>• Cooling system                             <ul style="list-style-type: none"> <li>Pumps</li> <li>Heat exchangers</li> <li>Piping</li> </ul> </li> <li>• Lubricating system                             <ul style="list-style-type: none"> <li>Heat exchangers</li> <li>Pumps</li> <li>Lube oil</li> </ul> </li> </ul>	<ul style="list-style-type: none"> <li>• Environmentally induced: dust, water, heat, oil, chemical, etc.</li> <li>• Maintenance errors: inadequate training, maladjustments, etc.</li> <li>• Fast starts and other regulatory induced factors</li> <li>• Design inadequacy, wrong application, or poor component</li> <li>• Operation induced: inadequate training and skills</li> <li>• Vibration induced</li> <li>• Fuel or lubrication degeneration</li> <li>• Gasket, seal, or organic material degeneration</li> <li>• Inadequate spares: quality, storage, ordering problems, date and specifications</li> <li>• Corrosion, oxidation</li> <li>• Thermal stress</li> <li>• Manufacturing or quality problems</li> <li>• Fatigue not related to vibration</li> <li>• Metal fatigue</li> <li>• Wear</li> <li>• Bacterial action</li> </ul>	<p><u>NRC Requirements</u></p> <ul style="list-style-type: none"> <li>• RG 1.9...surveillance, maintenance, periodic testing</li> <li>• RG 1.108...routine testing, maybe with drawn due to RG 1.9 revisions</li> <li>• General letter 83.41...“fast cold” starts</li> <li>• 10-CFR-50, appendix A, criterion 4, 5, 17, 18 &amp; 50 ... periodic testing</li> <li>• 10-CFR-5, appendix B, section XI requires... components &amp; system to perform satisfactory</li> <li>• 10-CFR-50, 55a Codes &amp; Srd - ASME BPVC section III, IX, IEEE STD 279</li> <li>• TS 3/4 8.1 surveillance/inspection for operation &amp; shutdown status</li> <li>• GSI B-56-improve reliability of Eng.</li> <li>• IEEE standard acceptable for use by NRC-RG 1.9</li> </ul>	<p>Integrated EDG program of testing, inspection, monitoring, trending and maintenance activities:</p> <ul style="list-style-type: none"> <li>• Testing/trending, change testing to a slower start test and acquisition of these testing parameters for trending</li> <li>• Improved inspection of weekly, monthly, and yearly to determine environmental stressors more effectively</li> <li>• Maintenance responsive to test &amp; inspection, e.g., do not over haul unless inspection and trends indicate the need.</li> <li>• Increased training for EDG Staff in on-site maintenance</li> <li>• System modifications to mitigate stressors</li> </ul>

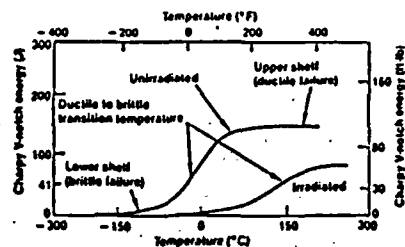
FIGURE 6.5. Understanding and Managing Aging of Emergency Diesel Generators

# Understanding and managing aging of PWR pressure vessels

- Materials**
- Vessel
    - Low alloy carbon steel - SA-533B-1, SA-508-2, SA-302 B
  - Cladding
    - Type 308 SS and 309 SS,
  - Weldments
    - Submerged arc (granular flux - Inde 88, 91, 124 and 1062 manganese-nickelboron nickel filler metal) narrow gap submerged arc, shielded metal arc, and electroslag
  - Closure studs
    - SA-540 Gc, B34 Class 3
- Stressors and Environment**
- Neutron flux and fluence, temperature, reactor coolant, cyclic thermal and mechanical loads, proteoids, and 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
Boltline region	Irradiation embrittlement <ul style="list-style-type: none"> <li>Chemical composition of vessel materials (Ca, Mn, P)</li> <li>Drop in upper shell energy (USE)</li> <li>SNH in reference nil-ductility-transition-temperature (RT<sub>NDT</sub>)</li> </ul> Environmental fatigue	<u>NRC Requirements</u> Surveillance program to assess irradiation damage, i.e., SNH in RT <sub>NDT</sub> and drop in USE (10 CFR 50 App. M, Reg. Guide 1.98, Rev. 2)  Pressurized thermal shock (PTS) screening criteria (10 CFR 50.55) PTS rule, RG 1.154 Damage evaluation (10 CFR 50 App. G) Pressure - Temperature (P-T) limits during startup, shutdown, artificiality, and inservice leakage and hydrostatic pressure test to prevent nonductile fracture (Tech. Spec. requirement, 10 CFR 50 App. G)  [P-T limits are also applied to non-boltline region]  Low temperature overpressurization (LTOP) protection setpoint (Tech. Spec. requirement)  Volumetric examination of all welds during each inspection interval (10 CFR 50.55a, FWB-2500, Reg. Guide 1.156, Rev. 1)  Flow evaluation (10 CFR 50.55a, FWB-3000)  Leakage and hydrostatic pressure tests (10 CFR 50.55a, FWB-5000)	<u>Recommendations</u> Include fracture toughness and tensile test specimens in surveillance program  Develop use of reconstituted and miniature specimens  Develop techniques for in site determination of mechanical properties  Perform accelerated irradiation tests of reconstituted specimens  Revise Reg. Guide 1.98, Rev. 2 to account for phosphorus with low copper  Use state-of-the-art ultrasonic inspection techniques for improved reliability of defect detection, sizing, and characterization <ul style="list-style-type: none"> <li>Automated amplitude-based systems</li> <li>Tip distraction techniques</li> <li>Large-diameter focused transducer</li> </ul> Use fatigue crack growth curves (ASME 8C 21, Appendix A)  Develop acoustic emission monitoring to detect crack growth (Nonmandatory appendix is being developed for ASME Section XI)	Neutron flux reduction  Inservice annealing (ASTM E 509-95) Determine effects of annealing and reembrittlement rate
Outlet/inlet nozzles	Environmental fatigue  Irradiation embrittlement Function of nozzle elevation Potential impact of (Reg. Guide 1.98, Rev. 2)	Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius sections during each inspection interval (FWB-2500)  Volumetric and surface examination of all dissimilar metal welds during each inspection interval (FWB-2500)	Use on-line fatigue monitoring (monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures)  Evaluate irradiation embrittlement damage	
Instrumentation nozzles CRDM housing nozzles	Environmental fatigue	Visual examination of external weld surface of 25% of nozzles during system hydrostatic test (FWB-2500)		
Closure studs	Environmental fatigue <ul style="list-style-type: none"> <li>preload cycles during head replacement</li> <li>Boric acid corrosion (if leakage occurs)</li> </ul>	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (FWB-2500)		

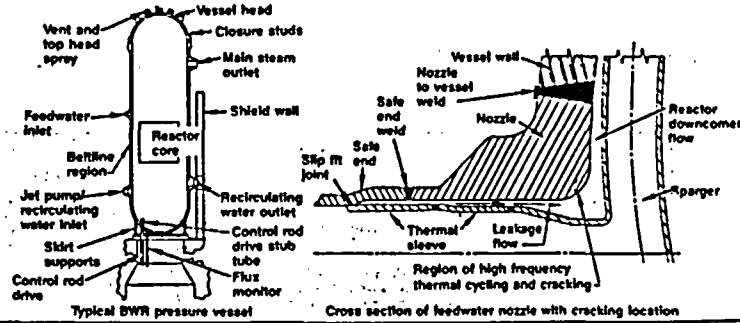
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FIGURE 6.6. Understanding and Managing Aging of PWR Pressure Vessels

# Understanding and managing aging of BWR pressure vessels

- Materials**
- Vessel**
    - Low alloy carbon steel SA-513B-1, SA-302B
    - Type 304 SS and 309 SS
  - Cladding** SA-508-2
  - Nozzles**
    - Type 304 SS, Type 316 SS, Inconel SB-164, Inconel SB-167, SA-508-1
  - Safe Ends**
    - Type 304 SS
  - Thermal Sleeves** Type 304 SS
  - Closure Studs** SA-540 Gr. B22 or B23
  - Weldments** SA-193 Gr. B7
- Stressors and Environment**
- Operational transients, neutron flux and fluence, temperature, reactor coolant and preloads



UNDERSTANDING AGING (Materials, Stressors, and Environmental Interactions)		MANAGING AGING		
Sites	Aging Concerns	Inservice Inspection, Surveillance, and Monitoring	Mitigation	
Feedwater nozzles and safe end welds	High-cycle thermal fatigue caused by feedwater leakage Environmental fatigue	<b>NRC Requirements</b>  Volumetric examination of all nozzle-to-vessel welds and nozzle inside radius sections during each inspection interval (IWB-2500)	<b>Recommendations</b> Use on-line fatigue monitoring (monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures) Develop criteria for assessing high-cycle fatigue damage	Modify design, revise operating procedures, and remove feedwater nozzle cladding to prevent fatigue cracking
Recirculation inlet/outlet nozzles and dissimilar metal welds	IGSCC crack initiated in HAZ may propagate into base metal Environmental fatigue	Volumetric and surface examination of all dissimilar metal welds during each inspection interval (IWB-2500)	Develop on-line corrosion monitoring Evaluate long-term effects of hydrogen water chemistry	Implement hydrogen water chemistry to reduce IGSCC damage
Welds - Control rod drive stub tubes - Interior attachments	IGSCC crack initiated in HAZ may propagate into base metal by corrosion and/or environmental fatigue	Visual examination of all accessible interior attachment welds during each inspection interval (IWB-2500)	Develop robotics system for remote inspection probe positioning and scanning	
Beltline Region	Irradiation embrittlement - Chemical composition of vessel materials (Cu, M, P) - Drop in upper shelf energy (USE) - Shift in reference ductility-transition temperature (RT <sub>NDT</sub> ) - Welds are more susceptible than base metal - Flux is lower than that in PWR vessel  Environmental fatigue	Surveillance program to assess shift in RT <sub>NDT</sub> and drop in USE (10 CFR 50 App. H, Reg. Guide 1.98, Rev. 2)  Damage evaluation (10 CFR 50 App. G) Pressure-temperature (P-T) limits during heatup, cooldown, criticality, and inservice leakage and hydrostatic pressure tests to prevent nonductile fracture (Tech. spec. requirement, 10 CFR 50 App. G.) [P-T limits are also applied to non-beltline region]  Volumetric examination of all shell welds during each inspection interval (10 CFR 50.55a, IWB-2500, Reg. Guide 1.150, Rev. 1)  Flaw evaluation (10 CFR 50.55a, IWB-3000)  Leakage and hydrostatic pressure tests (10 CFR 50.55a, IWA-5000, IWB-5000)	Revise Reg. Guide 1.98, Rev. 2 to account for phosphorous when copper content is low  Use state-of-the-art inspection techniques for improved reliability of defect detection, sizing, and characterization  Develop robotics system for remote inspection probe positioning and scanning  Include fracture toughness and tensile test specimens in surveillance program  Develop use of reconstituted and miniature specimens and accelerated irradiation of reconstituted specimens  Use fatigue crack growth curves (ASME Section XI, Appendix A)  Develop acoustic emission monitoring to detect crack growth (nonmandatory appendix is being developed by ASME Section XI)	Inservice annealing (ASTM E 509-96)  Determine effects of annealing and reembrittlement rate  Implement neutron flux reduction program
Closure Studs	Environmental fatigue - Preload cycles during head replacement Fretting	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval (IWB-2500)		
External attachment welds such as skirt supports	Low-cycle thermal and mechanical fatigue	Volumetric or surface examination (IWB-2500)		

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FIGURE 6.7. Understanding and Managing Aging of BWR Pressure Vessels

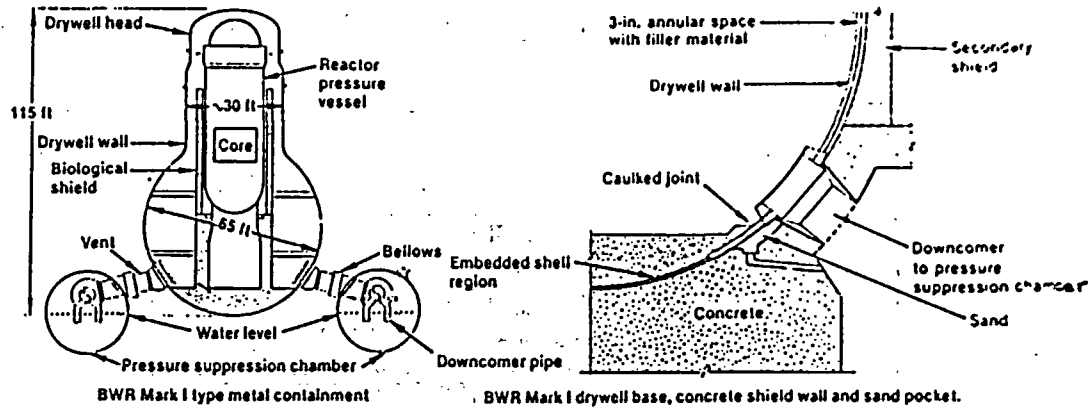
# Understanding and managing aging of BWR Mark I containments

**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, and leak tests



UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)			MANAGING AGING	
Sites		Aging Concerns	Inservice Inspections and Monitoring	Maintenance
Drywell	Exterior surface near sand pocket (unsealed gap)	Pitting, crevice corrosion, microbial influenced corrosion, and uniform corrosion	<p><b>NRC Requirements</b></p> <p>Leak tests (10 CFR 50 App. J)</p> <p><b>Recommendations</b></p> <p>Inservice Inspection-ASME subsection IWE</p> <ul style="list-style-type: none"> <li>- Being reviewed by NRC for inclusion in federal regulations</li> <li>- Visual examination of 25% of pressure retaining welds, and coated and uncoated surfaces during each inspection interval</li> </ul> <p>Visual examination of caulked joints at embedded regions</p> <p>Boroscopic examination of exterior surface near penetration</p> <p>Surface examination of bellows</p> <p>Monitor coolant leakage from faulty bellows</p> <p>Wall thickness measurements and trending</p> <p>Develop methods for inspecting metal surfaces through protective coatings</p> <ul style="list-style-type: none"> <li>- Magnetic particle testing</li> <li>- Eddy current testing</li> </ul> <p>Develop methods for inspecting drywell wall in the embedded shell region</p> <ul style="list-style-type: none"> <li>- Electromagnetic acoustic transducer</li> </ul>	<p><b>Recommendations</b></p> <p>Maintain surface coatings</p> <p>Check bellows alignment</p> <p>Maintain caulked joints at embedment region</p> <p>Evaluate use of cathodic protection for exterior surface of dry well</p>
	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	<p>Thermal, mechanical, and environmental fatigue</p>	
	Near waterline	Differential aeration		
	Below waterline	Microbial influenced corrosion		
Bellows	Heat-affected zone	Intergranular stress corrosion cracking	<p>Thermal and mechanical fatigue</p>	
	Cold-rolled portion	Transgranular stress corrosion cracking		

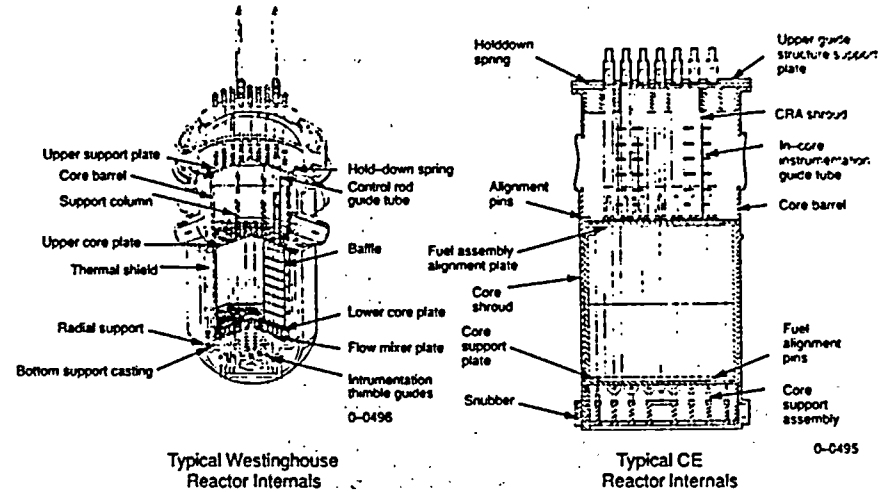
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FIGURE 6.8. Understanding and Managing Aging of BWR Mark 1 Containments

# Understanding and managing aging of PWR reactor internals

Materials	Instrument tubes (flux thimble tubes)	Type 304 SS
	Bolts and pins	Type 316 SS (W, except as noted below); Alloy A-286 (CE and B&W)
	CRGT split pins and radial support key bolts	Alloy X-750 (W)
	Thermal shield, core barrel, baffle and formers, upper and lower core support structures	Type 304 SS
	Flow mixer, cruciform instrument guides	Cast SS (Gr. CF-8) (some early W plants)
	Hold down spring	Type 403 SS (W and CE)
Stressors & Environment	Operational transients, temperature, fluid flow, residual stresses, bolt preloads, neutron radiation	



6.27

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Sites	Aging Concerns	Inservice Inspections, Surveillance, and Monitoring	Mitigation	
Instrument tubes (flux thimble tubes)	Fretting, wear, high-cycle fatigue	<b>NRC Requirements</b> Visual inspection of accessible subcomponents (ASME Section XI, Table IWB-2500-1, B14.10)  Preoperational vibration testing (USNRC RGs 1.20, 1.68)  Loose parts monitoring systems (USNRC RG 1.133)	<b>Recommendations</b> Develop ASME Code Section XI inspection requirements  Monitor the wear of W in-core instrument housings per USNRC Bulletin 88-09  Use advanced inservice inspection techniques for pins (e.g.; Framatome REBUS UT method for split pins)  Evaluate the use of cylindrically guided wave technique for bolts  Establish vibration monitoring programs using ex-core and/or in-core neutron noise detectors to detect excessive movements of core barrels and thermal shields  Use ANSI/ASME OM-5 procedure for neutron noise monitoring of core barrel axial preload	Use thicker-walled or double-walled thimble tubes (W plants)  Reduce preloads on high-strength bolts to mitigate IGSCC  Use improved heat treatments for Alloy X-750 high strength bolting materials to reduce IGSCC susceptibility  Remove thermal shields (or replace with neutron shield panels on exterior of core barrels) for designs that have experienced bolt failures  Use one-piece thermal shields rather than multi-piece designs (RPVs in some older W plants have inadequate access for one-piece installation)
Bolts and pins	IGSCC, high-cycle fatigue, stress relaxation			
CRGT split pins and radial support key bolts (W)	IGSCC			
Thermal shield	High-cycle fatigue, irradiation embrittlement, distortion			
Core barrel, baffles and formers	High-cycle fatigue, irradiation embrittlement			
Upper and lower core support structures	Irradiation embrittlement, corrosion-fatigue			
Flow mixer and cruciform instrument guides (some early W plants)	Thermal (ferrite phase) and irradiation (austenite phase) embrittlement			
Hold down spring (W and CE)	Stress relaxation			

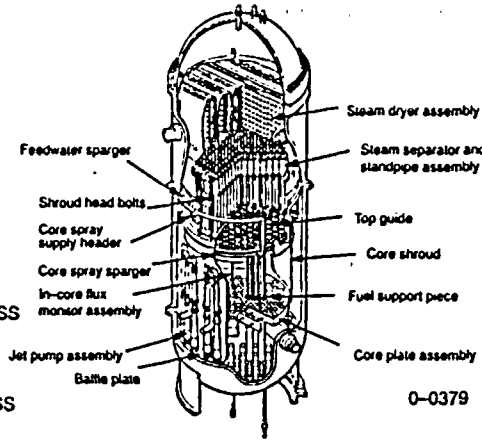
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FIGURE 6.9. Understanding and Managing Aging of PWR Reactor Internals

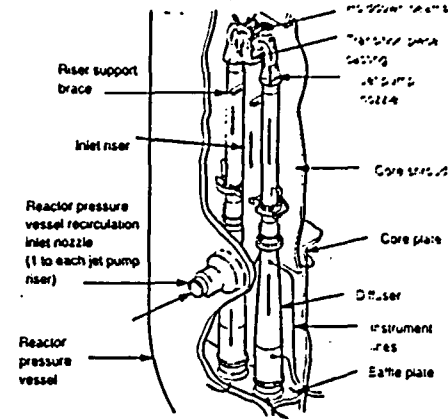
# Understanding and managing aging of BWR reactor internals

Materials	Attachment weld material	Alloy 182
	Core plate, core shroud	Type 304 SS
	Jet pump	
	holddown beams	Alloy X-750
	riser support braces	Type 304 SS
	welds	Type 308 SS
	castings	Gr. CF-8 SS
	Top guide	Type 304 SS
	Core spray and feedwater spargers	Type 304 SS
	Orificed fuel supports	Gr. CF-8 SS
	Peripheral fuel supports	Types 304 and 304L SS
	Baffle plate access hole covers, core support cylinder	Alloy 600
	Shroud head bolts	Alloy 600
	Neutron monitor dry tubes	Alloy 600, Type 304 SS

**Stressors & Environment** Temperature, residual stresses, flow-induced vibration, bolt preloads, dead weight, neutron and gamma radiation, chemical impurities and oxygen content in coolant



Typical BWR/3, BWR/4 Reactor Internals



Typical Jet Pump Assembly

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING	
Sites	Aging Concerns	Inservice Inspections, Surveillance, and Monitoring	
		Inspection/Requirement	Mitigation
Attachment welds to reactor vessel	IGSCC, corrosion-fatigue	<b>NRC Requirements</b>	Maintain strict control on coolant impurities to keep conductivity below 0.2 μS/cm Use hydrogen water chemistry to suppress dissolved oxygen in coolant from typical 200 ppb to below 20 ppb to mitigate IGSCC Assess effect of Hydrogen Water Chemistry on <ul style="list-style-type: none"> <li>- IASCC</li> <li>- fatigue crack growth</li> <li>- radiation fields</li> <li>- fuel performance</li> </ul> For replacement components <ul style="list-style-type: none"> <li>- Reduce preload stresses</li> <li>- Use high temperature annealing and age hardening of Alloy X-750 material</li> <li>- Remove crevices</li> </ul>
Core plate, core shroud	IGSCC, irradiation-assisted SCC (IASCC)	Visual inspection of accessible components (ASME Section XI, Table IWB-2500-1)	
Jet pumps	Holddown beams	Preoperational vibration testing (USNRC RGs 1.20, 1.68)	
	Riser support braces		
	Welds		
	Castings	Loose parts monitoring systems (USNRC RG 1.133)	
Top Guide	IASCC, IGSCC	Develop UT inservice inspection techniques for inaccessible attachment welds to reactor pressure vessel Use in-core SCC monitors	
Core spray and feedwater spargers	IGSCC, corrosion-fatigue		
Orificed fuel supports	IGSCC, Thermal and irradiation embrittlement, IASCC		
Peripheral fuel supports	IGSCC, IASCC		
Baffle plate access hole covers	IGSCC (crevices)		
Shroud head bolts	IGSCC (crevices)		
Neutron monitor dry tubes	IGSCC (crevices), IASCC		

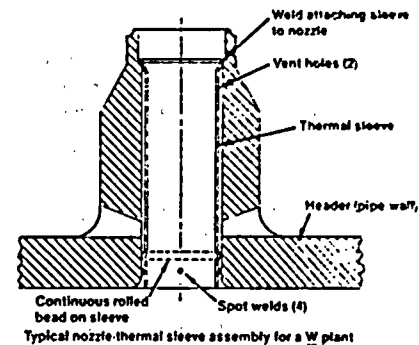
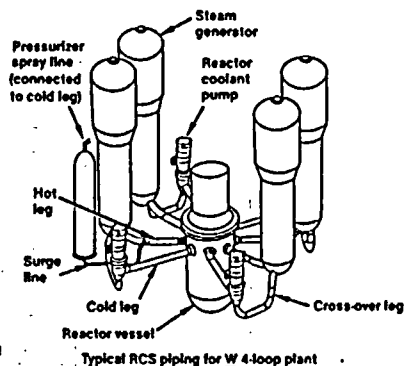
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FIGURE 6.10. Understanding and Managing Aging BWR Reactor Internals

# 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-316 Gr. 70 (CE), SA-106 Gr. C (B&W)
- Fittings**
- Statically cast SS - Gr. CF8A and CF8M (W); SA-316 Gr. 70, Type 309L SS (CE, B&W); Type 308L SS (B&W)
- Cladding**
- Type 308L SS (CE, B&W)
  - Type 304L SS (CE, B&W)
- Surge and spray lines**
- Type 316 SS, cast SS - Gr. CF8M (surge line in some CE plants)
- Charging, safety injection, and residual heat removal lines**
- Type 316 SS
- Nozzles on main coolant pipe**
- SA 105 Gr. 2 (CE), Type 304N SS (W)
- Safe ends**
- Type 316 SS, Inconel SB-168 (CE, B&W)
- Thermal sleeves**
- Inconel SB-168

**Stressors & Environment** Operational transients, temperature, flow induced vibrations, stratified flows, thermal stripping, valve leakage, and thermal shocks



6.29

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Sites	Aging Concerns	Inservice Inspections, Surveillance, and Monitoring	Mitigation	
<b>Nozzles and thermal sleeves</b> Charging Safety injection Surge Spray	Low- and high-cycle thermal and mechanical fatigue	<b>NRC requirements</b> Volumetric and surface examination of 25% of butt welds, including the following welds during each inspection interval (10 CFR 50.55a, IWB-2500): <ul style="list-style-type: none"> <li>All terminal ends in each pipe connected to vessels</li> <li>All dissimilar metal welds</li> <li>All welds having cumulative usage factor equal to or greater than 0.4</li> <li>All welds having primary plus secondary stress intensity range equal to or greater than 2.4 Sm</li> </ul> Same welds are required to be inspected during each inspection interval	<b>Recommendations</b> Perform more frequent examinations of nozzle welds and horizontal piping welds having significant fatigue damage Determine fatigue damage by on-line monitoring of pipe wall temperatures and coolant flows, temperatures, and pressures in nozzles and horizontal portions of piping subject to operational transients, including stratified flows and thermal sleeves 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: <ul style="list-style-type: none"> <li>Analytical modelling of inservice degradation</li> <li>Metallurgical evaluation to characterize microstructure</li> <li>NDE to establish correlation between ultrasonic attenuation and fracture toughness</li> </ul> Monitor valve leakage in safety injection and residual heat removal lines 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 Redesign surge and spray line piping by replacing short horizontal sections with sloped sections to prevent stratified flow conditions Redesign piping to eliminate deleterious effects of valve leakage Minimize valve leakage Maintain smaller temperature differences between pressurizer and hot leg coolants during heat up and cool down Maintain smaller temperature differences between the pressurizer and spray line coolants
Terminal end dissimilar metal welds (between carbon steel components and stainless steel piping)	Low- and high-cycle thermal and mechanical fatigue			
Surge and spray lines Charging, safety injection, and residual heat removal lines to first isolation valve	Low- and high-cycle thermal and mechanical fatigue			
Cast stainless steel piping Hot leg Cross-over leg Cold leg Fittings Surge line	Thermal embrittlement <ul style="list-style-type: none"> <li>Coolant temperature</li> <li>Ferrite content and spacing</li> <li>Chemical composition of base metal</li> </ul>	Flaw 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)		

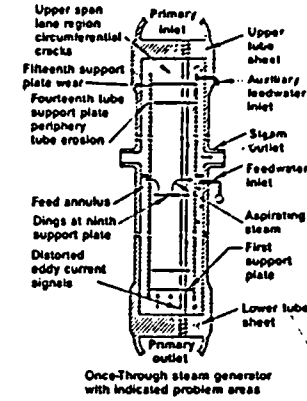
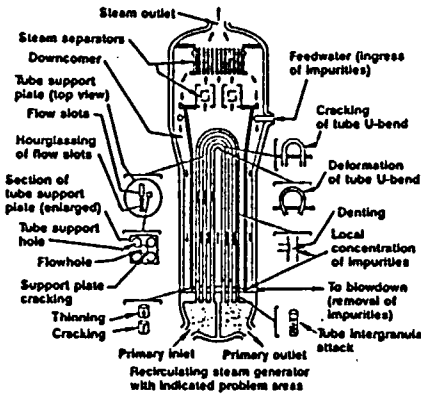
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FIGURE 6.11. Understanding and Managing Aging of PWR RCS Piping and Nozzles

# Understanding and managing aging of PWR steam generator tubes

- Materials**
- Tubes** - Inconel 600 or 690
  - Tube Sheet** - SA 508 clad with Ni-Cr-Fe alloy (equivalent to SS 168)
  - Tube Supports** - SA 285 Gr. C Ferritic SS Type 405 or 409
  - Sleeves** - Inconel 625 or nichel bonded on outside surface of Inconel 600 or 690
  - Plugs** - Inconel 600
- Steam Generator Types**
- Recirculating** - Westinghouse, Combustion Engineering
  - Once-Through** - Babcock & Wilcox
- Stressors and Environment**
- Residual stresses, primary coolant chemistry (primarily hydrogen concentration), secondary coolant chemistry (chlorides, oxygen, copper, sulfates), phosphate chemistry, resin leakage from condensate polishers, brackish water, temperature, flow-induced vibrations, flow-velocities, and operating transients



Understanding Aging (Materials, Stressors, and Environmental Interactions)			Managing Aging		
Types	Sites	Aging Concerns	Inservice Inspection, Surveillance, and Monitoring	Mitigation	
Recirculating Inside Surface	U-bends, roll transition, and dented regions	PWSCC (Pure water SCC) Tubes with low mill-annealing temperature are more susceptible	<b>NRCC Requirements</b>  Volumetric examination of hot leg side, U-bend portion, and (optionally) cold leg side of tubes in recirculating steam generators (TWB-2500)  Volumetric examination of the entire length of tubing in once-through steam generators (TWB-2500)  Frequency of inspection and number of tubes to be inspected (minimum of 3% of all tubes) are determined by Tech. Specs. (Reg. Guide 1.83)  Standards for allowable flaws in recirculating steam generators (standards for once-through steam generators are being prepared) (TWB-3521)  Flaw acceptance criteria determined by Tech Specs. (TWB-3630)  Criteria for determining necessity of plugging degraded tubes (Reg. Guide 1.121)	<b>Recommendations</b>  Follow Steam Generator Owners Group's guidelines for continuous monitoring and control of secondary water chemistry  Reduce uncertainties in inspection results and quantify flaw growth rates  Monitor field performance of various sleeve designs  Perform inservice inspection of tube plugs	Prevent transient conditions in secondary water chemistry, install filters between condensate polishers and steam generators. Use ultrafiltration of makeup water and remedy condenser leakage as quickly as possible  Use shotpeening and rollopeening to introduce compressive residual stresses on tube inner surface in the roll transition region, and anneal U-bends to reduce PWSCC  Apply nickel plating on the inner surface of the tubes to prevent PWSCC crack initiation and propagation  Use tube rolling to eliminate tube sheet crevices and use crevice flushing, crevice alkalinity neutralization, alkaline impurity control, acid chloride elimination, hot soaks, sludge lancing, pressure pulse, water slap, chemical cleaning, and boric acid additions to control IGA/IGSCC  Eliminate copper pickup by use of titanium or stainless steel condenser tubes, and replace the copper-bearing alloys in the feedwater train to reduce pitting and denting  Use pH-volatile treatment water chemistry, sludge lancing, chemical cleaning, hot soaks, hot blowdown and flushing, and elimination of hideout chemical concentration to control wastage  Use chemistry control to prevent concentration of impurities leading to fatigue crack initiation in once-through steam generators  Use lane-flow blocker in once-through steam generators to mitigate environmental fatigue
	Tube plugs				
Recirculating Outside Surface	Hot-leg tubes in tube-to-tubesheet crevice region	IGSCC, IGA			
	Cold leg side in sludge pile or where scale containing copper deposits is found	Pitting			
	Tubes in tube support regions	Denting			
	Inadequately supported tube if dented near the top support plate	High-cycle fatigue			
	Contact points between tube and antivibration bar	Fretting			
	Tubes above tubesheet	Wastage			
Once-Through Outside Surface	Tubes	Erosion-corrosion Fatigue			
	Tubes in upper tubesheet region	Environmental fatigue			

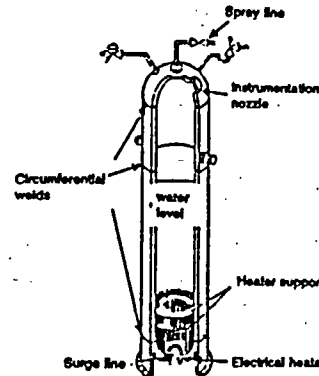
FIGURE 6.12. Understanding and Managing Aging of PWR Steam Generator Tubes



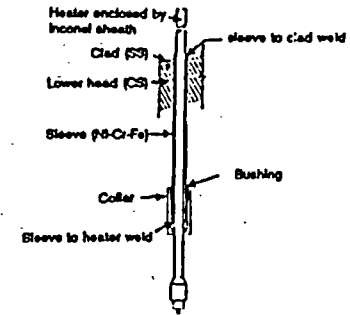
# Understanding and managing aging of pressurizer, surge and spray lines

Materials	Vessel	<ul style="list-style-type: none"> <li>shell, A-533, Gr. B, Class I</li> <li>Cladding, Type 304 SS &amp; Ni-Cr-Fe Alloy</li> </ul>
	Heater	<ul style="list-style-type: none"> <li>Sheath, Inconel Insulation MgO</li> </ul>
	Fittings	<ul style="list-style-type: none"> <li>Statically cast SS - Gr. CF8A and CF8M (W); SA 518 Gr. 70, Type 309L SS (CE, BAW); Type 308L SS (BAW)</li> </ul>
	Cladding	<ul style="list-style-type: none"> <li>Type 308L SS (CE), Type 304L SS (BAW)</li> </ul>
	Surge line	<ul style="list-style-type: none"> <li>Type 316 SS, cast SS-Gr. CF8M some (CE plants)</li> </ul>
	Spray line	<ul style="list-style-type: none"> <li>Type 316 SS</li> </ul>
	Nozzles on main coolant pipe	<ul style="list-style-type: none"> <li>SA 105 Gr. 2 (CE), Type 304 N SS (W)</li> </ul>
	Thermal sleeve	<ul style="list-style-type: none"> <li>Inconel SB-108</li> </ul>

Stressors and Environment: Operational transients, temperature, flow induced vibrations, stratified flows, thermal stripping, thermal shocks, heater mechanical wear and element burnout and erosion and corrosion



Typical W pressurizer and connections



Typical CE heater equipment

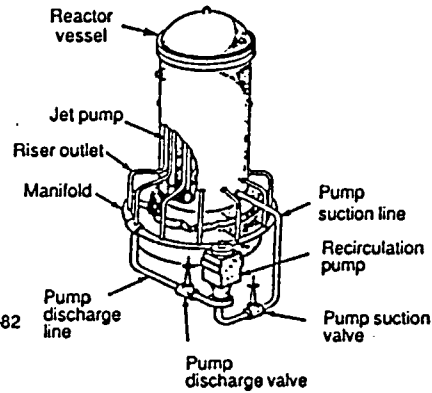
UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Site	Aging Concerns	Inservice Inspections, Surveillance, and Monitoring	Mitigation	
Nozzles and thermal sleeves • Instrumentation • Surge • Spray	Low and High-cycle thermal  Erosion	<u>NRC requirements</u> Volumetric and surface examination of 25% of butt welds including the following welds each inspection interval 10 CFR 50.55a, IWB-2500: <ul style="list-style-type: none"> <li>All dissimilar metal welds</li> <li>All welds having cumulative usage factor equal to or greater than 0.4</li> <li>All welds having stress intensity range of <math>2.4 S_m</math></li> </ul> Same welds are required to be inspected during each inspection interval	<u>Recommendations</u> Perform more frequent examination of nozzle welds having high cumulative usage factor.  Determine fatigue damage by on-line monitoring of coolant and piping temperatures, 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 improved NDE method to detect crack growth in the base metal and welds.  Monitor valve leakage  Develop UT to detect flaws on 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  Preventive or predictive maintenance for heater replacement  Use improved stem packing materials
Terminal end dissimilar metal welds (between carbon steel components and stainless steel piping)	Low-Cycle thermal and mechanical fatigue	Flow detection and evaluation 10 CFR 50.55a, IWB-3000  Leakage Hydrostatic pressure tests 10 CFR 50.55a, IWB-5000  ASME Section III, NB-3210 and ASME Section XI, ISI  Cycle counting of specified design transients		
Cast stainless steel piping • Surge line • Spray line • Valves • Fittings	Low-cycle thermal and mechanical fatigue Thermal embrittlement Erosion spray valve Boric acid Corrosion Stem packing/wear and degradation with age and service life Before degradation	Tech Spec's requirements <ul style="list-style-type: none"> <li>Cycle counting of specified design transients</li> <li>Leakage rates</li> <li><math>\Delta T</math> limits for heatup/cooldown</li> </ul>		
Heater sheath failures	Small LOCA via heater element and heater sleeve			
Vessel wall	High and low cycle thermal fatigue			

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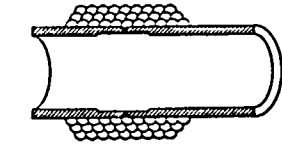
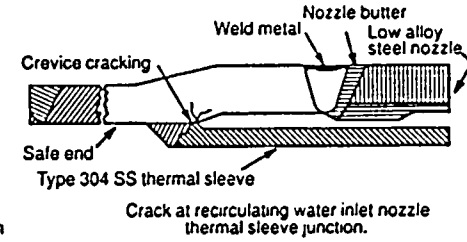
FIGURE 6.13. Understanding and Managing Aging of Pressurizer, Surge and Spray Lines

## Understanding and managing aging of BWR recirculation piping

<b>Materials</b>	<b>Piping</b>	- Types 304, 316L, and 316NG SS
	<b>Fittings, pumps and valves</b>	- Statically cast SS - Gr. CF8 and CF8M
	<b>Safe ends</b>	- Types 304, 316, 316NG SS and Alloy 600
	<b>Pipe-to-pipe and pipe-to-elbow welds</b>	- Type 308L SS
	<b>Safe end-to-nozzle weld</b>	
	<b>Weld material</b>	- 308L/308 SS
	<b>Nozzle butter</b>	- 308L SS
	<b>Safe end butter</b>	- None
		or
		- Alloy I-182/1-82
		- Alloy I-182
		- None or Alloy I-182



BWR recirculation piping system.



Typical weld overlay reinforcement.

**Stressors & Environment** Operational transients, residual tensile stresses, applied stresses, oxygenated coolant, impurities in coolant, and contaminants on the outside surface.

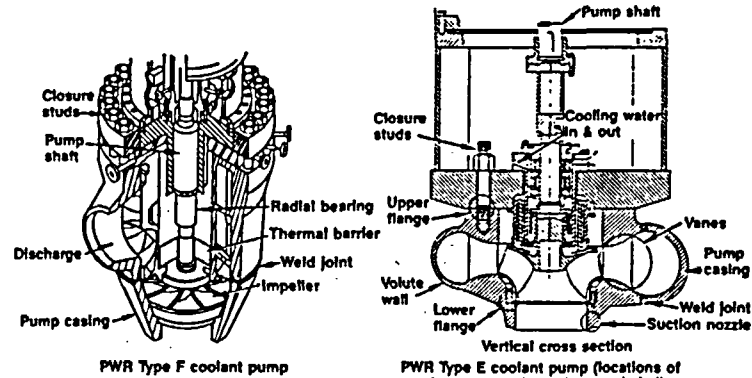
UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Sites	Aging Concerns	Inservice Inspections, Surveillance and Monitoring	Mitigation	
Weld sensitized heat-affected zone, Furnace sensitized safe end, bimetallic and trimetallic welds (Types 304, 304L, 316, 316L SS)	IGSCC - Inside surface - Outside surface, if high residual tensile stresses (may be introduced, for example, by stress improvement) and contaminants are present	<b>NRC Requirements</b> Ultrasonic inspection of weldments as per NRC Generic Letter 88-01, Reduced inspection period and increased sample size for piping made of non-IGSCC resistant materials (e.g., Type 304 SS)	<b>Recommendations</b> Develop improved methods to inspect weld overlay repairs.  For leak detection: - Install moisture sensitive tape on susceptible region of piping. - Use acoustic emission monitoring method	Employ weld overlay, mechanical stress improvement, induction heating stress improvement, and clamping device for repairing welded piping
Sites with crevices and small amount of cold work	IGSCC	- For weldments without crack, depends on when stress improvement implemented	- Conduct on-line monitoring of coolant chemistry	Use Type 316NG SS, corrosion-resistant cladding, solution heat treatment, and heat sink welding for replacement piping
Sites with highly stressed regions and large amount of cold work (Type 316NG SS)	Transgranular stress corrosion cracking (TGSCC)	- For weldments with cracks, depends on whether cracks have been reinforced by weld overlay or mitigated by stress improvement	For effective Hydrogen Water Chemistry control: - Maintain electrochemical corrosion potential on standard hydrogen electrode scale to $\leq -230$ mV. - Maintain coolant conductivity $\leq 0.3$ $\mu$ S/cm	Evaluate the use of 347NG SS as piping material
Sites subject to cyclic stresses	Fatigue	- Inspection procedure for weld overlay repair should be capable of detecting cracks that were 75% of the original wall thickness		Assess the effects of Hydrogen Water Chemistry on overall plant operation - radiation fields - fuel performance
Duplex (austenitic-ferritic) stainless steels - Fittings, pump and valve castings	Thermal embrittlement of high delta ferrite (>10% ferrite) regions, IGSCC of low delta ferrite (<10% ferrite) regions	- Inspection period can be as short as one refueling cycle, and sample size can be as large as all welds		- acceptable length and frequency of short outages of hydrogen water chemistry

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FIGURE 6.14. Understanding and Managing Aging of BWR Recirculation Piping

# Understanding and managing aging of LWR coolant pumps

Materials	Casing	<ul style="list-style-type: none"> <li>Statically cast SS Gr. CF8, CF8A, CF8M, or SA-508 with authentic SS clad</li> </ul>
	Closure Studs	<ul style="list-style-type: none"> <li>SA-193 Gr. B7 or SA-540 Gr. B23</li> </ul>
Pump Types	Cover Gasket	<ul style="list-style-type: none"> <li>Type 304 SS flexituff (stainless steel-graphite-asbestos material)</li> </ul>
	Shaft	<ul style="list-style-type: none"> <li>Type 304 or 316 SS</li> </ul>
Stressors and Environment	Westinghouse	<ul style="list-style-type: none"> <li>Type F</li> </ul>
	Babcock & Wilcox	<ul style="list-style-type: none"> <li>Type F, Type E</li> </ul>
	Combustion Engineering	<ul style="list-style-type: none"> <li>Type F, Type E</li> </ul>
	General Electric	<ul style="list-style-type: none"> <li>Type C</li> </ul>
Stressors and Environment		Temperature, operating transients, residual stresses, gasket leakage, alternating bending stresses, turbulent mixing of hot and cold coolant, and vibration



Understanding Aging (Materials, Stressors, and Environmental Interactions)		Managing Aging			
	Sites	Aging Concerns	Inservice Inspection, Surveillance, and Monitoring	Mitigation	
Casing	Cast SS base metal	Thermal embrittlement <ul style="list-style-type: none"> <li>Coolant temperature</li> <li>Ferrite content and spacing</li> <li>Chemical composition</li> </ul> Fatigue at high stress intensity locations	<b>NRC Requirements</b>  Volumetric examination of all welds and visual examination of internal surfaces, in at least one pump casing during each inspection interval (10 CFR 50.55a, IWB-2500)	<b>Recommendations</b>  Characterize ferrite distribution and existing flaws in pump casing and welds  Develop standards for allowable flaw sizes for pump casing base metal  Develop techniques for monitoring actual degree of thermal embrittlement in pump casing  Perform examinations of high stress regions	
	Low-alloy base metal	Fatigue at high stress intensity locations			
	Welds	Fatigue at welds having high residual stresses or high stress intensities  Thermal embrittlement			
	Welds with low ferrite content	IGSCC if sensitized and lower fatigue strength if microvoids are present			
Closure Studs		Boric acid corrosion caused by gasket leakage	<b>NRC Requirements</b>  Volumetric examination of all closure studs (on pump being examined) during each inspection interval (10 CFR 50.55a, IWB-2500)	Include visual examinations in inservice inspection requirements to detect corrosion wastage  Use cylindrically guided wave technique to detect both corrosion wastage and cracks	
Shaft	Sites with high stress concentrations or residual stresses	High-cycle mechanical fatigue caused by alternating bending stresses			Evaluate use of modified cylindrically guided wave technique for shaft inspection
	Near thermal barrier	High-cycle thermal fatigue caused by turbulent mixing			
	Chrome-plated shafts	Cracks in chrome plating may propagate in shaft by high-cycle mechanical fatigue			
Use improved cover gaskets with better spring-back characteristics, proper gasket installation, and cleanliness control to prevent boric acid corrosion of closure studs  Leave leak-off lines between inner and outer gaskets unplugged  Install instruments on leak-off lines to detect leakage of reactor coolant  Remove chrome plating on pump shafts					

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FIGURE 6.15. Understanding and Managing Aging of LWR Coolant Pumps

- Cast Stainless Steel - An ANL investigation of the microstructural characteristics of cast stainless steel from the Shippingport Station primary system check valves has contributed to clarification of the thermal embrittlement processes that can occur at LWR operating temperatures. The phase changes that had occurred in this naturally-aged material were similar to those observed in artificially-aged laboratory specimens providing validation of aging projections based on the extrapolation of laboratory data.
- Neutron Shield Tank Samples - Mechanical property measurements were conducted on samples of the inner wall of the Shippingport Station neutron shield tank. Preliminary results of an ANL evaluation of these samples (including base metal and weld materials exposed to different neutron flux levels) suggest that the transition temperature changes resulting from low-temperature low-flux irradiation are less severe than a previous study had indicated. Differences with the previous study appear to be caused by neutron spectral differences and not to flux rate effects.
- Inverter/Battery Charger - Naturally-aged inverters and battery chargers from the Shippingport Station were tested by BNL under the NPAR Program. Although some aging-induced changes were noted, it was concluded that aging had not substantially affected equipment operation.
- Nuclear Protection System Panel - An INEL evaluation of a naturally-aged 1960s vintage nuclear protection system panel and rack showed that the system was operating close to the original response time of the equipment when it was new.
- Check Valves - An ORNL evaluation of a piston-lift check valve from the Shippingport Station found significantly more wear than would be expected based on the valve's normal service environment.
- Motor-Operated Valves - An 8-in. diameter gate valve and operator from the Shippingport Station were refurbished and requalified at INEL. No operational problems had been observed during periodic testing of the valve during Shippingport Station operations. The refurbished valve and operator were subsequently tested as part of an internationally sponsored seismic research program. The structural integrity of the valve and operator were not affected by the seismic excitations. However, the studies did reveal a previously unrecognized cable sizing problem

that resulted in the issuance of NRC Information Notice No. 89-11: "Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cable Sizing."

Radiological Assessment - A PNL assessment of the corrosion film on primary coolant piping samples from the Shippingport Station disclosed comparatively low concentrations of long-lived activation products and very low concentrations of fission products and transuranic radionuclides, reflecting the high integrity of the fuel cladding during reactor operation.

A list of publications from the Shippingport studies is provided in Appendix E.

#### 6.5.2.4 Regulatory Instrument Review

Under the NPAR Program, a single-source document has been generated (Werry 1990) that delineates the regulatory documents that govern the inspection and monitoring of reactor pressure vessels, steam generators, pressurizers, and major reactor piping, for use in both aging studies and the LR process.

### 6.6 NPAR SUPPORT TO LICENSE RENEWAL

The NPAR technical bases were developed principally to address aging issues for plant operation up to 40 years. However, these technical bases have been valuable to NRC staff in the analysis of issues related to LR. Utilization of NPAR results in development of the basis for LR is addressed in Section 5.0. Research that contributes to resolution of technical safety issues could be accommodated in the NPAR Program if issues warrant further investigation.

## 7.0 NPAR REVIEW AND INTEGRATION ACTIVITIES

The Nuclear Plant Aging Research (NPAR) Program has been the subject of a series of integration and review activities; these activities are summarized in this section.

### 7.1 TECHNICAL INTEGRATION REVIEW GROUP FOR AGING AND LIFE EXTENSION

Prompted by a November 1985 memorandum from the U.S. Nuclear Regulatory Commission (NRC) Executive Director for Operations, the Technical Integration Review Group for Aging and Life Extension (TIRGALEX) panel was organized in 1986, with L. C. Shao as chairman, involving representatives from each NRC office. An early activity involved a survey of NRC programs with relevance to nuclear power plant (NPP) aging. The Review Group identified and documented technical safety and regulatory policy issues associated with NPP aging and license renewal (LR), reviewed relevant NRC and external programs, and provided recommendations for future NRC actions. A report was published summarizing TIRGALEX work, Plan for Integration of Aging and Life-Extension Activities (NRC 1987).

### 7.2 AGING AND LIFE EXTENSION COORDINATING COMMITTEE

Following completion of the TIRGALEX initiative, further coordination and critique of aging-related studies in NRC was carried out under the Aging and Life Extension Coordinating Committee (ALEXCC) in 1987 and 1988. The general consensus of these critiques was that the aging program was a vital element in developing both the LR Rule and the Maintenance Rule.

### 7.3 NPAR RESEARCH REVIEW GROUP

Before the formal organization of the review group, the NPAR Program Managers' Technical Review Meeting was held February 24-26, 1987 at Oak Ridge National Laboratory. A major NRC-wide NPAR review session took place in 1988, with organization of the Review Group, drawing from the NRC Offices and the Regions. After an extended effort, the organization was completed in August 1988. The first major activity was a 3-day Review Group meeting in Rockville, Maryland, March 21-23, 1989, where the ongoing NPAR research at the laboratories was reviewed. The Review Group fully endorsed the NPAR Program. The second review by the Review Group was held in Bethesda, Maryland, March 20, 21, 1990. The third review was held in Rockville, Maryland, March 26-28, 1991.

#### 7.4 OTHER NPAR REVIEW ACTIVITIES

In addition to these extensive review activities, the NPAR Program or elements of the Program have been reviewed as follows:

- In October 1987, the NPAR Program was reviewed by the Senior Research Program Steering Group on Component Integrity. The NPAR Program was approved as presented.
- In November 1987, the NPAR Program was reviewed by the Nuclear Utility Plant Life Extension (NUPLEX) Working Group of the Nuclear Utility Management and Resources Council (NUMARC). The presentation was received as being well conceived and productive.
- In December 1987, the NPAR Program was presented to the Equipment Quantification Advisory Group (EQAG) of the Electric Power Research Institute (EPRI) meeting in Charleston, South Carolina. General comments were made to the effect that the NPAR Program was generating data on aging that would be very useful to the industry.
- In January 1988, an NPAR in-depth briefing for NRC's Office of Analysis and Evaluation of Operational Data personnel was held. Overall evaluation was that the program was proceeding to develop needed aging data.
- In February 1988, Electrical and Mechanical Engineering Branch staff and contractors participated in the EPRI/Industry Workshop on Power Plant Cable Condition Monitoring and discussed the NPAR cable studies. Industry strongly supports and gave high priority for these studies.
- In March 1988, the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Reliability reviewed the NPAR work on motor operated valves and check valves. The ACRS Subcommittee endorsed the need for continuing this research effort.
- In March 1988, Commissioner Rogers, his staff, and the staff of the other Commissioners, were given an in-depth briefing on the NPAR Program. He and his staff voiced strong approval of the program and the need to maintain it, and if possible, to accelerate it.
- In August 1988, the NRC conducted the highly successful International Nuclear Power Plant Aging Symposium. Many NPAR papers were presented; also papers from the industry and from foreign speakers. For the first time, NUMARC and EPRI attendees raised the question as to the "correct" definition of aging.
- In September 1988, at a coordination meeting with the newly formed NUPLEX group, the NPAR Program was again presented. Though there

was a general consensus that work being done was very worthwhile, the issue of the "correct" definition of aging was again raised. No consensus was reached.

- In November 1988, the NPAR Program was reviewed by the Nuclear Safety Research Review (NSRR) Committee. The NSRR generally endorsed the NPAR Program.
- In April 1989, a meeting was held with NUPLEX to exchange information on NRC's aging program and steps being taken in the development of nuclear plant licence renewal criteria.
- In May 1990, the ACRS reviewed progress on the NPAR Program.

In addition to the review activities listed above, various elements of the NPAR Program have been reviewed and critiqued on at least five occasions by the members of ALEXCC before it was replaced by the NPAR Research Review Group.

It should also be noted that as part of the cooperative effort developed between NPAR and EQAG, NPAR draft reports are reviewed by members of EQAG, and the industry insights thus gained, where compatible with the NRC's safety mission, are incorporated into the final published report.

Another coordinated activity is organization of a committee to develop common aging terminology. The committee members were selected from EPRI NRC/NPAR, NPAR laboratory staff, NUMARC, and utilities. The effort was funded by EPRI and was coordinated by MPR Associates. A draft report containing the recommended definitions has been circulated for committee comment; the comments are currently being resolved. The draft will be issued for wider industry and regulatory review before it is published.



## 8.0 COORDINATION WITH OTHER PROGRAMS, INSTITUTIONS, AND ORGANIZATIONS

Various institutions and industry organizations have performed studies and instituted programs related to aging research. Several ongoing programs are also producing significant results that cannot and should not be duplicated. Therefore, a major emphasis in the Nuclear Plant Aging Research (NPAR) Program Plan is that proper coordination and integration of research on plant aging are achieved at various levels. This approach will help achieve overall program goals and objectives and ensure the efficient use of available resources.

Ongoing Nuclear Regulatory Commission (NRC) programs related to aging and license renewal (LR) are being conducted by the Office for Analysis and Evaluation of Operational Data (AEOD), Office of Nuclear Reactor Regulation (NRR), and Office of Nuclear Regulatory Research. The NPAR Program coordination and technical integration with other agency programs are in place or will be implemented, facilitated by inputs from the NPAR Review Group. In addition to NRC-sponsored research, aging and LR programs are being sponsored in the U.S. by nuclear industry groups including Electric Power Research Institute (EPRI), nuclear steam supply system (NSSS) vendors, utilities, architect engineers, and the Department of Energy (DOE). Also nuclear plant aging and extended operation programs are being conducted in a number of foreign countries. Where the scopes of these programs are available, they are summarized in Appendix C.

The International Atomic Energy Agency (IAEA) in Vienna, Austria, has organized a program on nuclear power plant (NPP) aging. A major symposium, held in Vienna in June 1987, attracted approximately 140 representatives from 30 countries. Other IAEA activities include an advisory group to assist in planning activities, a consultant group on recordkeeping, a consultant meeting on "Safety Aspects of Nuclear Power Plant Aging (November 1989), and a technical committee meeting on "The Evaluation and Management of the Safety Impact on Nuclear Power Plant Aging" (November 1989). Additional meetings were held in June and November of 1990. An IAEA coordinated research program on NPP aging and life extension has been organized and is conducting pilot studies on four key safety-related components (see Appendix C).

Figure 1.1 illustrates the coordination with ongoing NRC programs, and with external programs, involving both domestic and foreign organizations. It is very likely that new aging-related programs will be sponsored in the future by outside organizations. An important and continuing activity in the NPAR Program is identifying new projects and establishing appropriate interfaces.

A listing of the more relevant programs, external to the NPAR Program, is given in Table 8.1. Appendix C contains descriptions of ongoing aging-related programs.

**TABLE 8.1. Selected Programs Relevant to NPAR Aging Studies and Programs**

Office of Nuclear Regulatory Research

- 10 CFR Part 54-LRR - To develop and publish a License Renewal Rule (LRR). 10 CFR Part 51 - To develop and publish an environmental rule for License Renewal (LR).
- Numerous projects addressing generic safety issues and specific SSCs have aging and LR implications. In most cases, however, aging and extended operation are not the dominant considerations in the work.

Office of Nuclear Reactor Regulation

- Maintenance and Surveillance Program--Maintenance and surveillance are key mechanisms for controlling aging. Phase I of the program has been completed, and Phase II is under way. The work addresses many other safety and regulatory issues in addition to aging and LR.
- Safety System Functional Inspection Program--The objective of this program is to ensure that plant modifications have not impaired the effectiveness of selected safety systems. This includes the detection, control, and prevention of selected safety system degradation, whatever the cause; age-related degradation is covered along with other casual factors not related to aging.
- Safety System Outage Modifications Inspection Program--The objective is to verify that modifications and repairs are conducted properly so that safety is not compromised. This covers modifications and repairs that may be aimed at managing aging or permitting LR.
- Electrical Distribution Inspection Program--the objective is to verify that the electrical distribution system still meets the original or modified design and to ensure component performance meets requirements.

Office of Analysis and Evaluation of Operational Data

- NPRDS Analyses--AEOD analyzes data for the Nuclear Plant Reliability Data System (NPRDS), producing statistical and engineering evaluations of component failure modes, time to failure, operating conditions that affect failure, and chemical and physical conditions affecting component-wearout rates.
- Licensee Event Report Database and Analyses--AEOD has developed the Sequence Coding and Search System (SCSS), which can be used in

studies of component life and component aging. SCSS is a comprehensive, computerized LER-based system with features that are highly useful in aging studies, e.g., cause codes specifically identifying age-related degradation.

NRC REGIONS

- Field Inspection and Maintenance--Dissemination of NPAR report summaries to NRC regional inspection personnel is currently being explored. The format of these report summaries will be tailored to provide concise information of direct value to the inspectors in the conduct of their activities.

EPRI/DOE/INDUSTRY

- EPRI/DOE/Industry Cooperative Program--This cooperative program is addressing the full spectrum of policy, safety, regulatory, economic, and technical issues associated with extended operation. The BWR and PWR pilot LR studies are complete. Development of lead plant LR requests are underway, targeted for 1991. Specific LR-related programs sponsored separately by DOE/SNL and EPRI are summarized in Appendix C.

CODES AND STANDARDS COMMITTEES

- ASME committees for operation and maintenance(see Table 6.2) are developing standards that address aging management of key safety-related equipment, with input from NPAR in selected topics.
- ASME Section XI--A Special Working Group on Plant Life Extension meets quarterly. They have active participation from EPRI, NRC, DOE, NSSS suppliers, and utilities. ASME XI is just beginning to develop code changes based on LR considerations.
- IEEE Working Group 3.4, under the IEEE Nuclear Power Engineering Committee, is developing recommendations for changes to IEEE standards based on aging and extended operation considerations.
- IEEE Working Group 3.3, under the IEEE Nuclear Power Engineering Committee, is developing recommended methods for the industry to mitigate equipment aging and to preserve equipment operability.

FOREIGN COUNTRIES

- Several countries, including Canada, France, Germany, Japan, Spain and Taiwan have programs that address aging and extended operation.

## 9.0 SCHEDULES AND RESOURCE REQUIREMENTS

The current schedule for the Nuclear Plant Aging research (NPAR) program has been adjusted to support the regulatory activities anticipated for nuclear plant license renewal (LR). The estimated schedules and milestones for completing specific research activities for aging assessments and LR are provided in this section. These general schedules, and particularly schedules for evaluating specific systems, structures, and components (SSCs), depend on funding, assignment of priority, and degree of coordination and participation by other institutions and organizations. The NRC/RES staff and its contractors are actively pursuing participation from domestic and foreign institutions and organizations. Requirements for LR will have increased priority in NPAR resource allocations.

The schedule and resource requirements reflect the need for early identification and prioritization of the technical safety issues, based on their risk significance. Other major elements affecting these requirements are the availability of topical reports from industry programs, the work scope required to resolve essential safety issues, and the development of proposed regulatory review procedures.

The schedules for major elements of the NPAR Program are given in Figures 9.1 to 9.4. The scheduling of the Phase-I and Phase-II assessments for components and systems is shown for each major activity. Also shown are the schedules for utilizing the research results, completion periods, and delays caused by budget cuts or reprioritization.

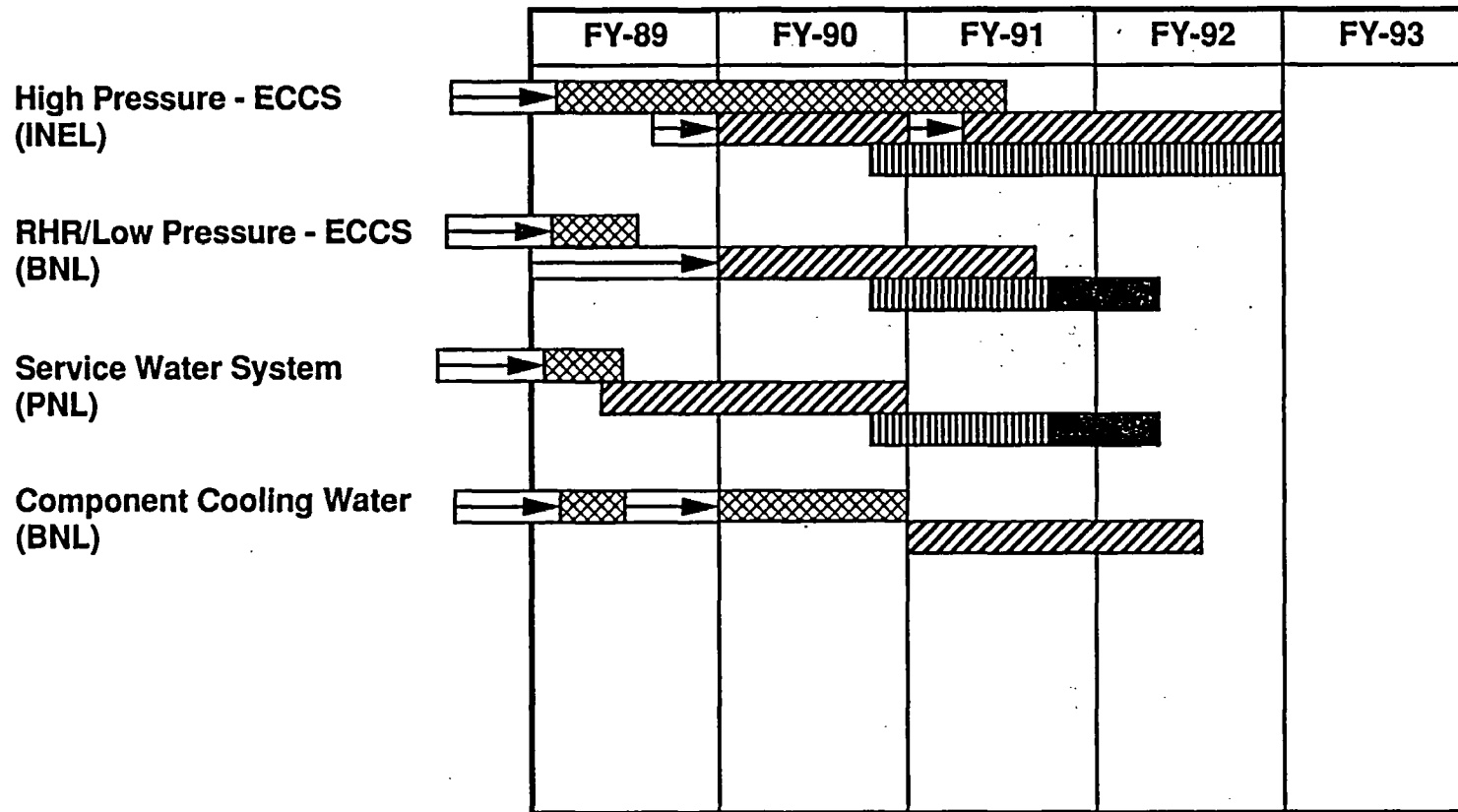
Table 9.1 summarizes the planned NPAR accomplishments for fiscal years 1991 and 1992. The schedules and major activities are based on the current NPAR research priorities. Note that the activities and schedules can change as information is developed in the program and as additional inputs are provided and program needs are identified.

Note also that the number of systems and components and the degree and depth of assessments and analyses that can be conducted effectively will depend on the availability of funds and the period of time over which the results are required.

Participation will be needed from NRC staff at an average level of five full-time professionals per year for the duration of the program. When participation by outside organizations is achieved, the resources provided from outside NRC will be identified in future revisions of this plan.

# NPAR Milestones and Schedules--Systems

9.2



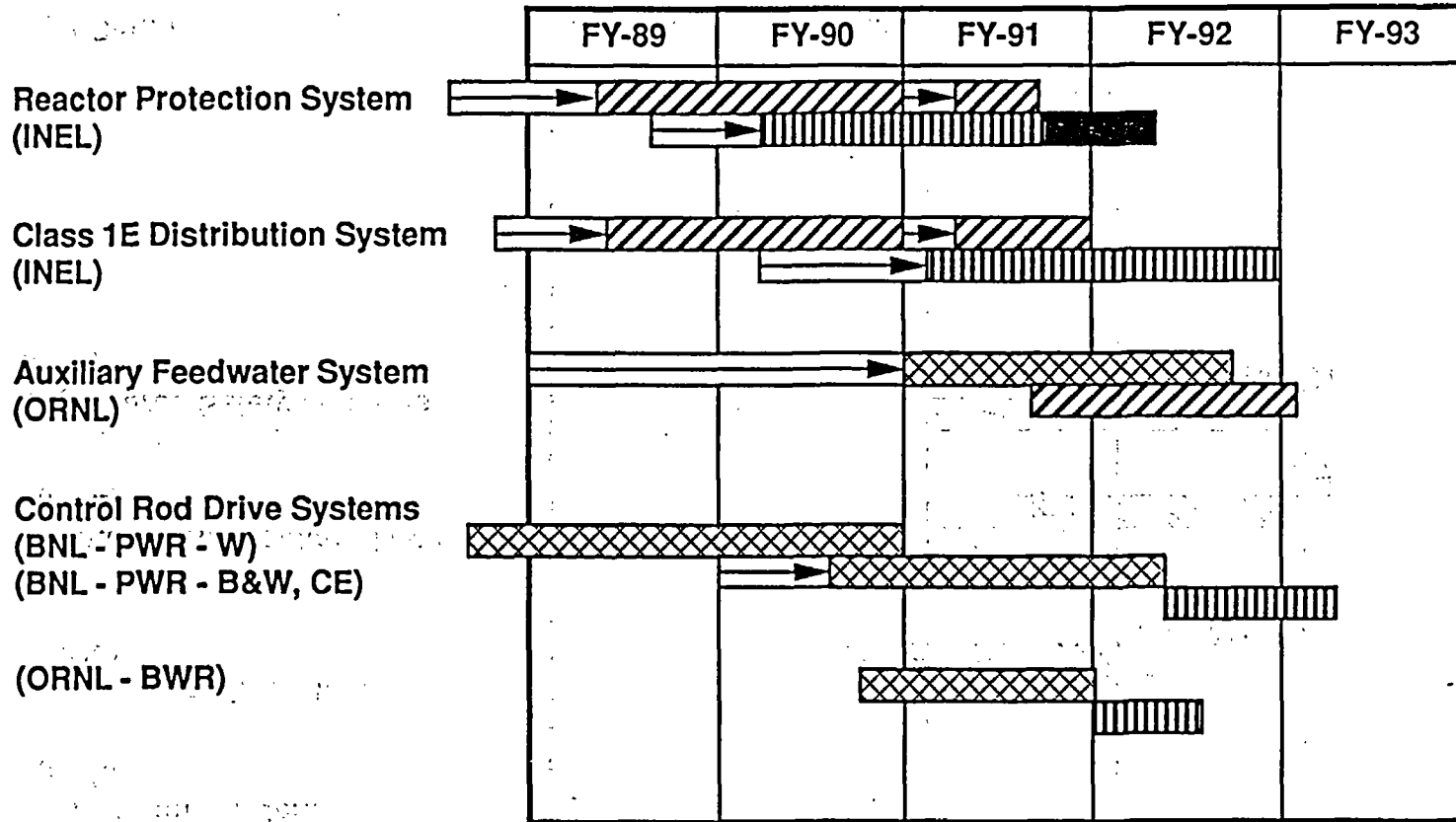
Complete    
  Phase 1    
  Phase 2    
  Utilization of Research Results  
 Signifies Delays Due to Budget Cuts or Reprioritization

39103033.1

FIGURE 9.1. NPAR Milestones and Schedules--Systems

# NPAR Milestones and Schedules--Systems

9.3



Complete
  Phase 1
  Phase 2
  Utilization of Research Results

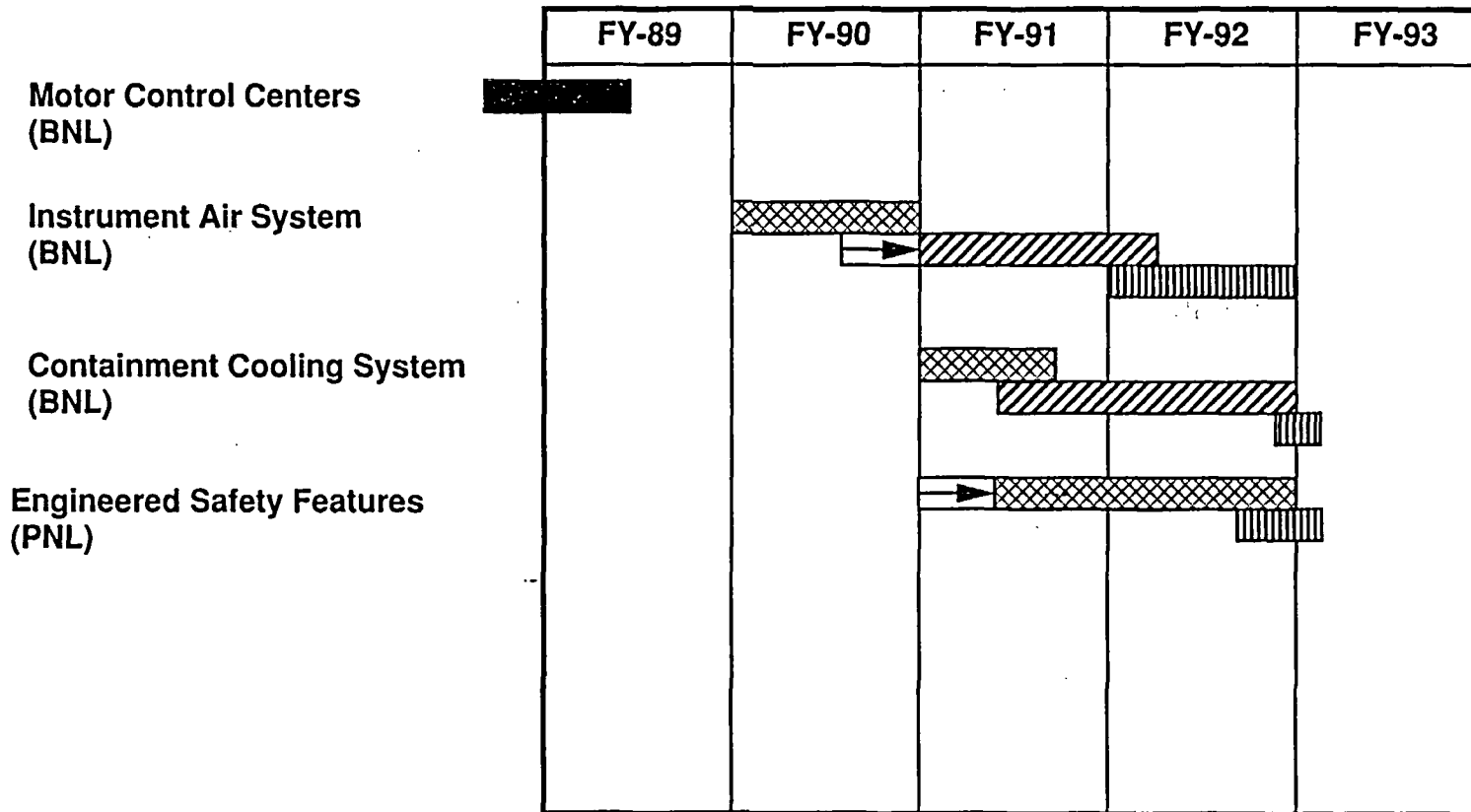
Signifies Delays Due to Budget Cuts or Reprioritization

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FIGURE 9.1. (contd)

# NPAR Milestones and Schedules--Systems

9.4



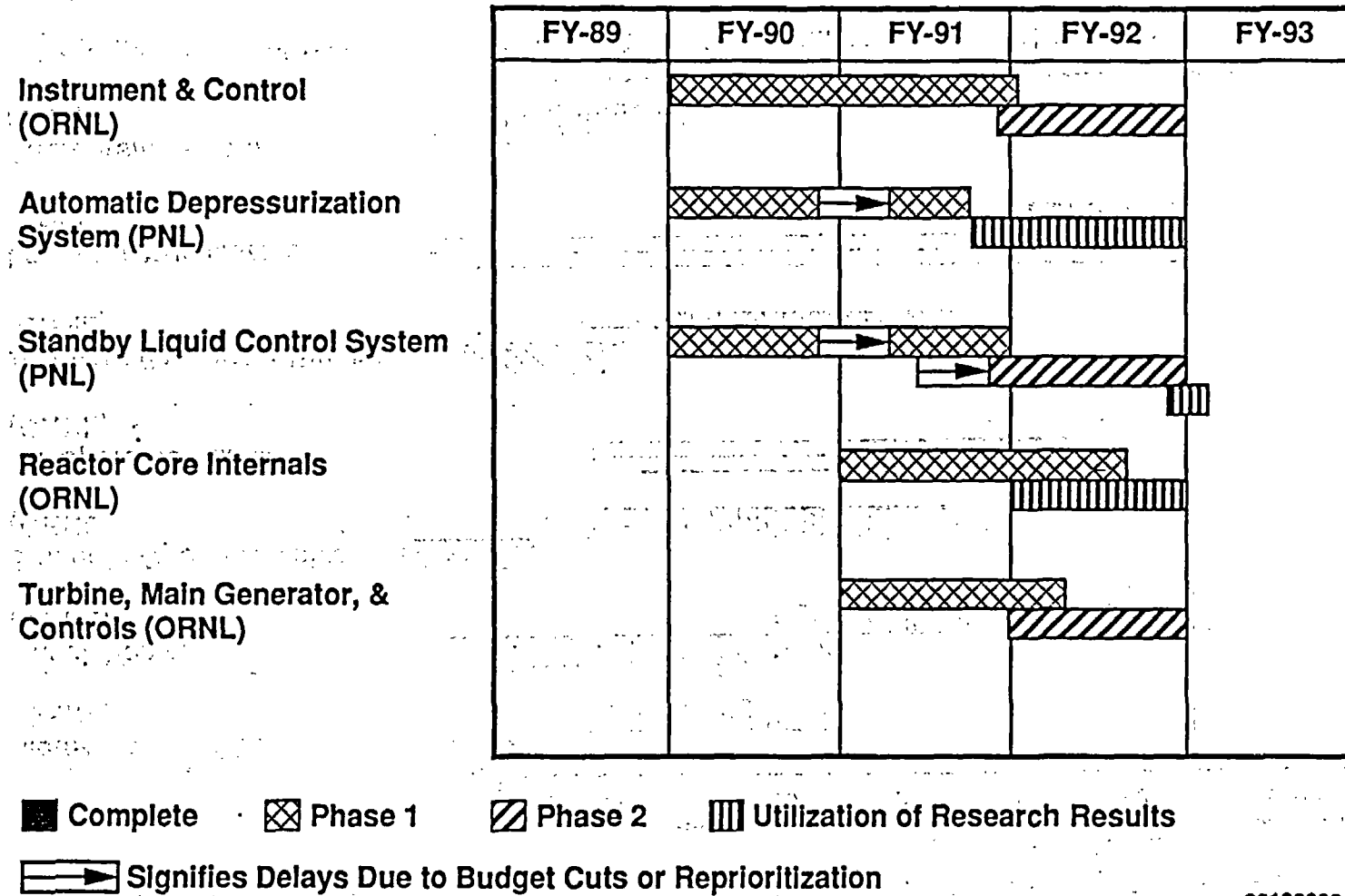
Complete    
  Phase 1    
  Phase 2    
  Utilization of Research Results  
  Signifies Delays Due to Budget Cuts or Reprioritization

39103033.3

FIGURE 9.1. (contd)

# NPAR Milestones and Schedules--Systems

9.5

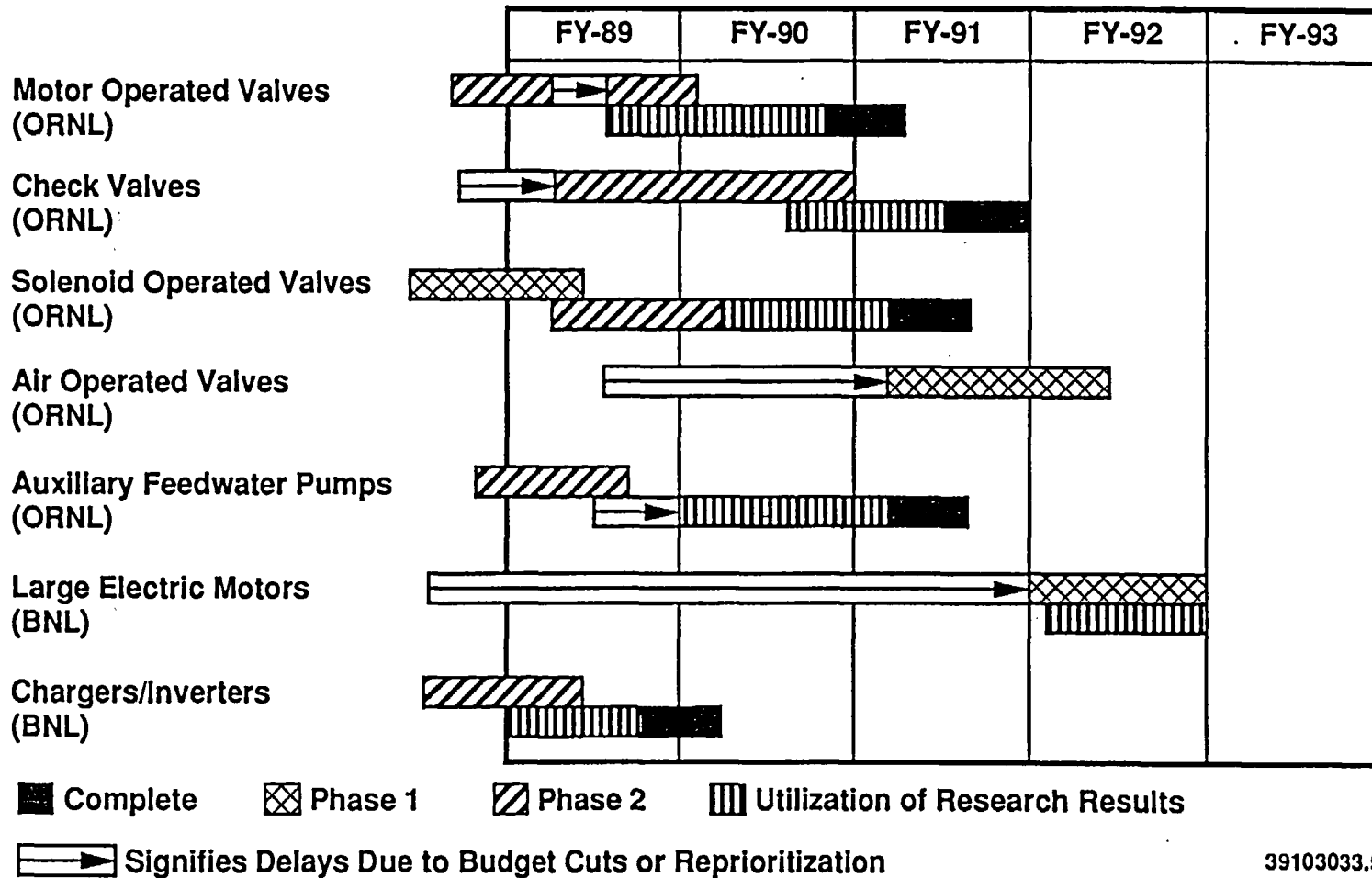


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FIGURE 9.1. (contd)

# NPAR Milestones and Schedules--Components

9.6



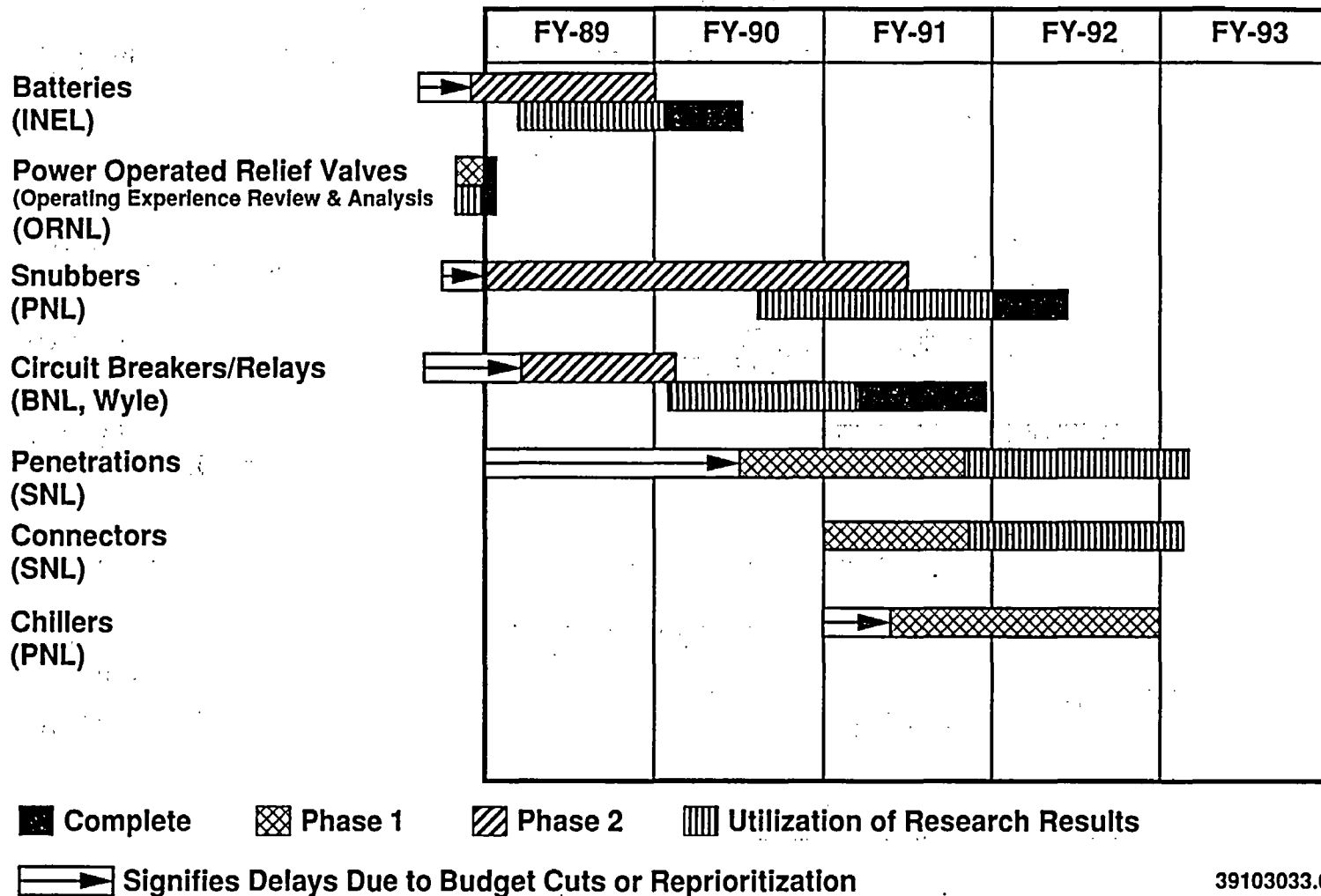
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FIGURE 9.2. NPAR Milestones and Schedules--Components



# NPAR Milestones and Schedules--Components

9.7

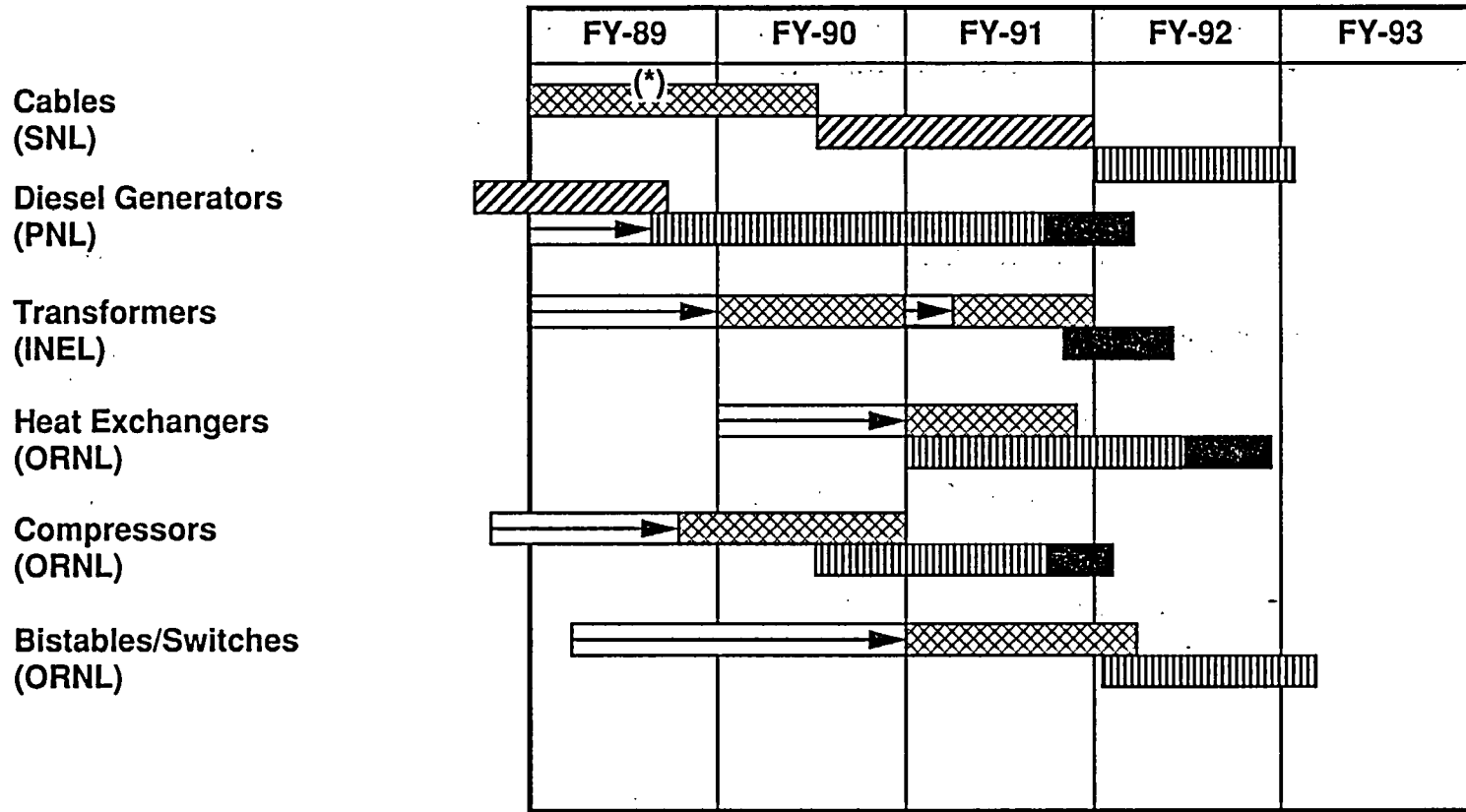


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FIGURE 9.2. (contd)

# NPAR Milestones and Schedules--Components

9.6



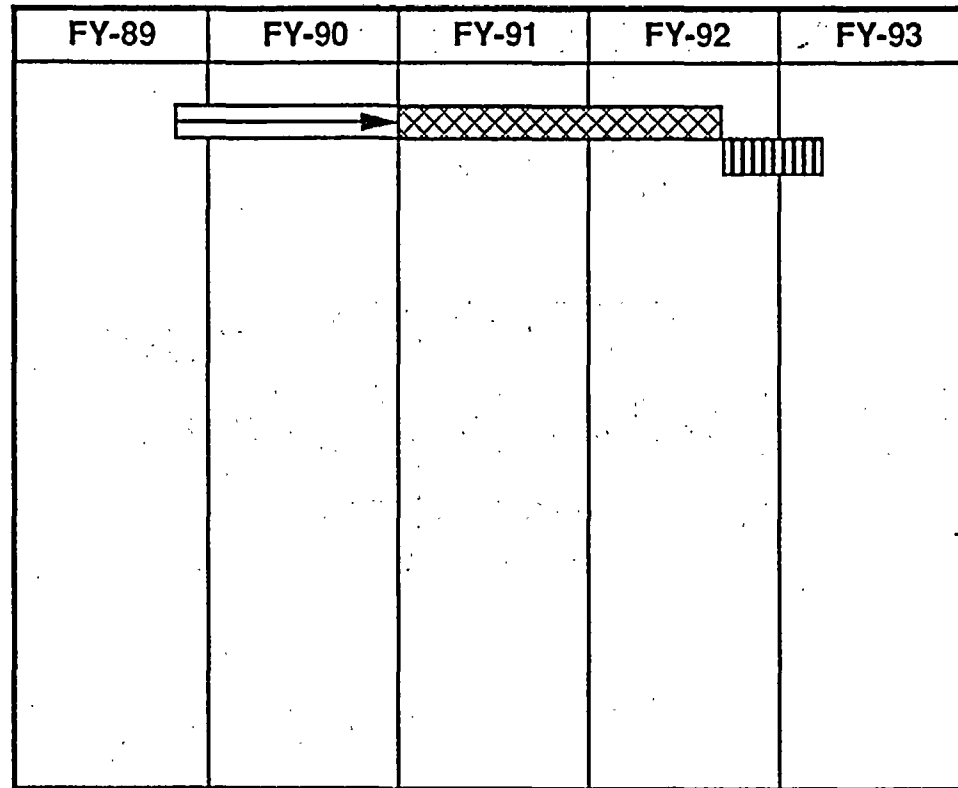
Complete    
  Phase 1    
  Phase 2    
  Utilization of Research Results  
  Signifies Delays Due to Budget Cuts or Reprioritization  
 (\*) Carried on from EQ Program

39103033.7

FIGURE 9.2. (contd)

# NPAR Milestones and Schedules--Components

Main Steam Isolation Valves (ORNL)



Complete    
  Phase 1    
  Phase 2    
  Utilization of Research Results

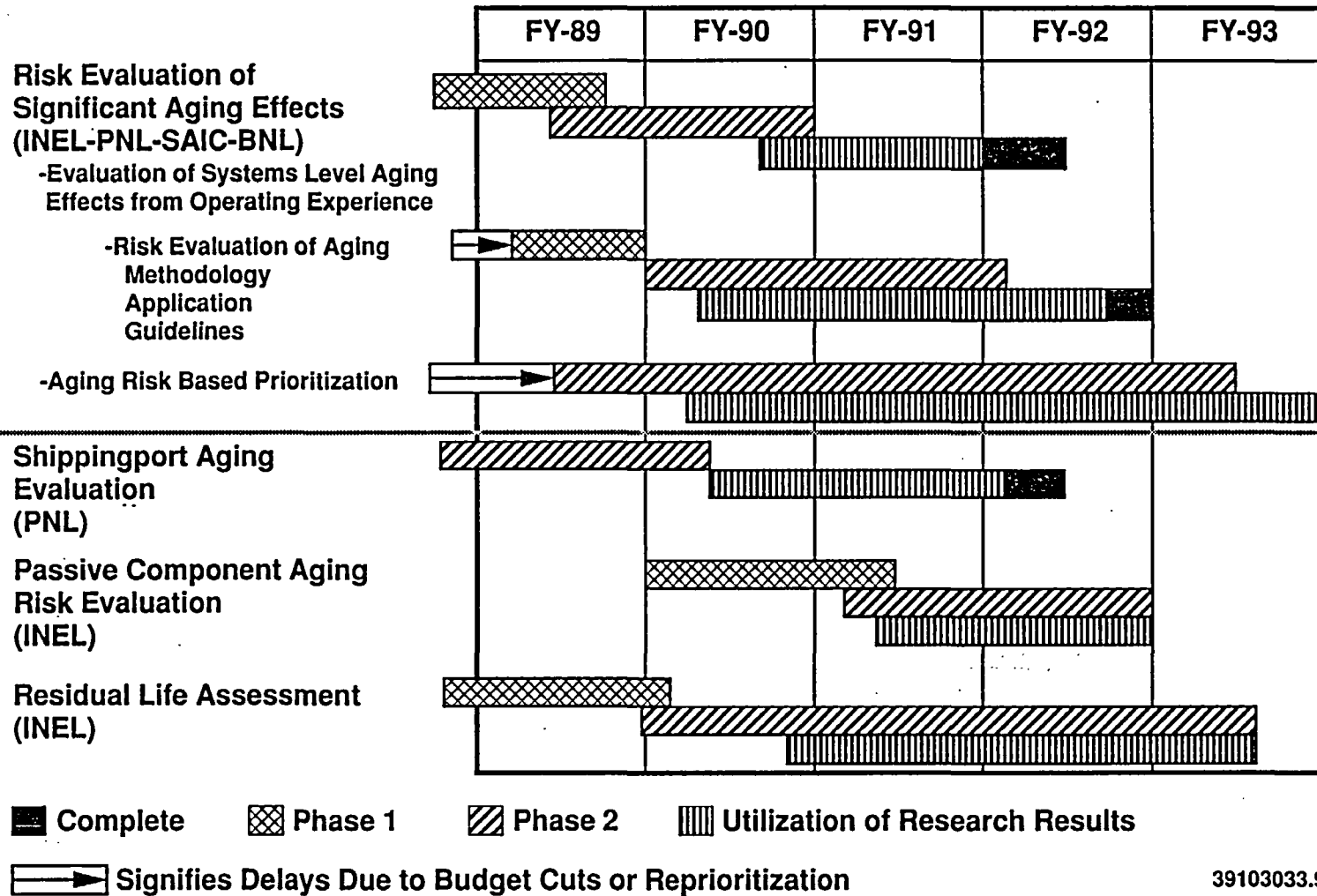
Signifies Delays Due to Budget Cuts or Reprioritization

39103033.8

FIGURE 9.2. (contd)

# NPAR Milestones and Schedules--Special Topics

9.10



39103033.9

FIGURE 9.3. NPAR Milestones and Schedules--Special Topics

# NPAR Milestones and Schedules-- Aging Program and Issues Coordination

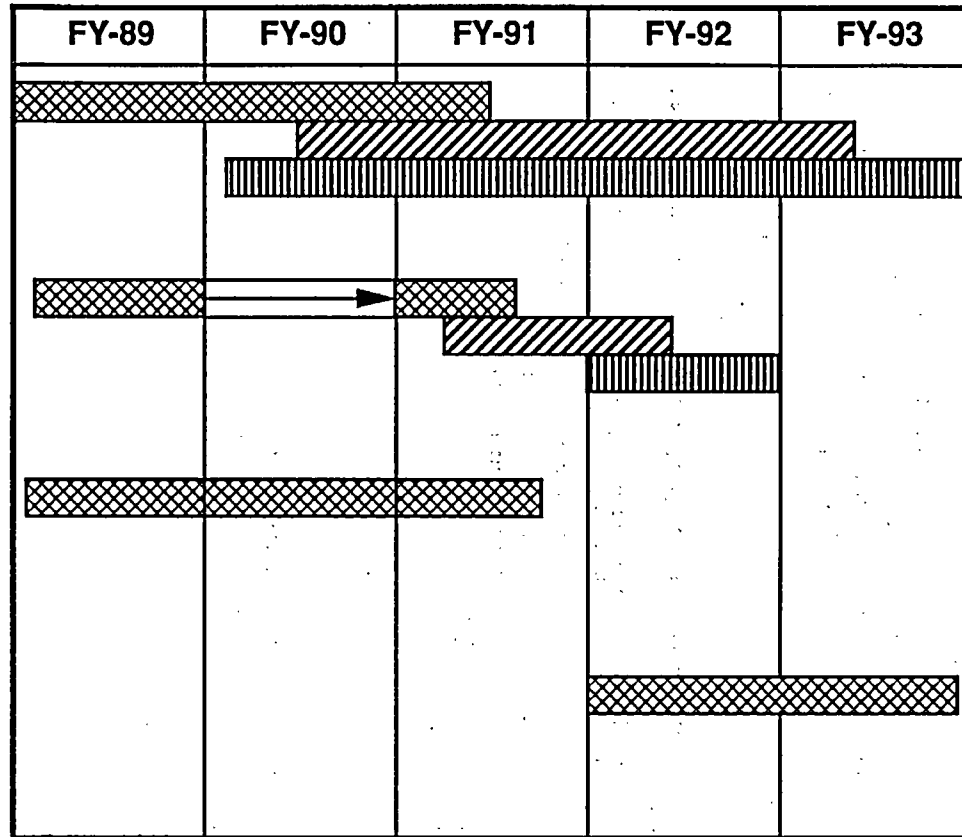
9.11

Technical Bases for  
Developing Regulatory  
Guides and  
Review Procedures

Recommendations for  
Technical Specifications  
Modifications from Aging  
Viewpoint

Recommendations for  
Data and Recordkeeping  
Modifications from Aging  
Viewpoint

Aging of Mothballed NPPs



■ Complete    ▣ Phase 1    ▤ Phase 2    ▥ Utilization of Research Results

⇨ Signifies Delays Due to Budget Cuts or Reprioritization

39103033.10

FIGURE 9.4. NPAR Milestones and Schedules--Aging Program and Issues Coordination

**TABLE 9.1. Planned NPAR Accomplishments for FY 1991 to 93**

- Issue a draft Master Regulatory Guide (Format and Content) to address technical issues relating to aging degradation for license renewal.
- Issue a manual on management of aging.
- Develop guidelines for recordkeeping and good maintenance practices for license renewal, if required.
- Develop a step-by-step procedure for treatment of aging in probabilistic risk assessment.
- Develop a procedure to account for testing and maintenance, influencing aging degradation, in PRA.
- Complete the regulatory instrument review of major LWR components from aging perspective and for license renewal consideration.
  - Containment Basement
  - Cables
  - Selected Pumps and Valves
- Complete the review of the technical specifications from aging perspective and generate recommendations to Office of Nuclear Reactor Regulation (NRR).
- Complete the reviews of industry (NUMARC) technical reports; assist NRR in developing staff technical positions; assist NRR in developing/modifying Standard Review Plans for license renewal considerations.
- Complete the development of model(s) for residual-life assessment of selected major LWR components and structures.
- Complete assessment of selected emerging inspection, monitoring, and material testing techniques.
- Complete Phase-II NPAR studies on the following components and systems:
  - Residual Heat Removal/Low Pressure System
  - Reactor Protection System
  - Class 1E Distribution System
  - Instrument Air System
  - Snubbers
  - Cables
  - Fire Susceptibility/Electrical Components (complete FY 1993)
- Complete Phase-I NPAR studies on the following components and systems:
  - Control Rod Drive Systems
  - BOP System Aging Effects
  - Automatic Depressurization System
  - Reactor Core Internals
  - Air Operated Valves
  - Large Electric Motors
  - Penetrations
  - Connectors
  - Chillers
  - Transformers
  - Heat Exchangers
  - Bistables/Switches
  - Main Steam Isolation Valves
  - Engineered Safety Features
- Complete Phase-I and initial Phase-II NPAR studies on the following systems and components:
  - High Pressure-ECCS
  - Auxiliary Feedwater System
  - Containment Cooling System
  - Instrument and Control
  - Standby Liquid Control System
  - Turbine, Main Generator and Controls
- Apply NPAR lessons learned to advanced LWRs.

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APPENDIX A

NPAR PROGRAM PARTICIPANTS

## APPENDIX A

### NPAR PROGRAM PARTICIPANTS

This appendix contains a brief summary of the NRC contractors functioning under the Nuclear Plant Aging Research (NPAR) Program. The research is currently being conducted by the following U.S. Department of Energy (DOE) national laboratories:

- Brookhaven National Laboratory (BNL), Long Island, New York
- Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho
- Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee
- Pacific Northwest Laboratory (PNL), Richland, Washington
- Sandia National Laboratory (SNL), Albuquerque, New Mexico.

Also supporting and participating in this research are the National Institute of Science and Technology (NIST), Scientific Applications International Corporation (SAIC), and Wyle Laboratories. In addition, work is subcontracted to other private engineering firms and private and academic consultants when specialized expertise is required. The laboratories frequently interact with utilities operating nuclear power plants in conducting and validating the research on aging of components and systems. Tables 3.1, 3.2 and 4.1 summarize the specific roles of the laboratories and the schedules for completion of the individual work elements.

The major format for documentation of the results of the NPAR aging assessments is NUREG/CR reports. Some studies are documented in technical reports from the individual laboratories. Other results are reported in proceedings of meetings and symposia, in proceedings of the annual NRC Water Reactor Safety Information meeting, and in peer-reviewed journal articles.

Kondic and Hill (1990) summarized the 72 NPAR reports published through May 1990. Since then, approximately 20 more NPAR reports have been issued in final or draft form.

APPENDIX B

UNDERSTANDING AND MANAGING AGING



## APPENDIX B

### UNDERSTANDING AND MANAGING AGING

In this appendix, the nature and processes of aging in nuclear power plants (NPPs) are reviewed. Aging processes are described, including degradation mechanisms. Also identified are methods to manage aging in NPPs. Finally, the potential impact of aging on NPP safety is explored.

#### B.1 NATURE OF AGING PROCESSES

From the regulatory perspective, the term "aging" can denote either favorable<sup>(a)</sup> or unfavorable changes in properties. As an NPAR working definition, age-related degradation is defined as the cumulative degradation occurring within a reactor system, structure, or component, which, if unmitigated, may result in loss of function and impairment of safety. Aging is a complex process that begins as soon as a component or structure is produced and continues throughout its service life. Aging plays a significant role in the operation of a nuclear plant and must be factored into the determination of safe operating lifetime limits. It also is important in evaluations for license renewal. Each nuclear plant, including those still under construction or being mothballed, should be systematically evaluated for the effects of aging. Factors that can cause aging include the following:

- Natural internal chemical or physical processes - Examples include the embrittlement of metals and plastics because of structural or chemical changes, high-temperature sensitization causing susceptibility to intergranular stress corrosion cracking in austenitic stainless steels, and oxidation or cross linking of polymers with a resultant loss in toughness and dielectric strength.
- External stressors and environments - Irradiation, primary and secondary coolant chemistry, and vibratory loads are the typical examples of stressors introduced by the operating environment. Freezing and thawing, brackish water, and humidity are typical examples of stressors introduced by external environment. Synergistic influences of electrical and mechanical stressors in combination with other internal and external environments also contribute to degradation processes.

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(a) Certain aging processes, such as curing of concrete, can lead to improved properties.

- Service wear - Service wear of rotating equipment and wear of the drive assembly in a control rod drive mechanism are typical examples.
- Excessive testing - Frequent fast-start testing of emergency diesel generators is a prominent example.
- Improper installation, application, or maintenance - During Phase I of the NRC Maintenance and Surveillance Program, (NUREG-1212) the investigators concluded that 48% of the licensee event reports (LERs) were maintenance related, and that over 60% of the forced outages were due to component failures.

These factors and others can act either singly or synergistically to degrade a system, structure, or component.

## B.2 UNDERSTANDING AGING

The effective management of aging begins with an understanding of the aging processes. The requisite understanding may be either empirical or mechanistic, depending on the nature and potential consequences of a particular aging process.

### B.2.1 Degradation Mechanisms

Stressors and environments act in concert on materials to cause age-related degradation. Many mechanisms can potentially contribute to deterioration processes. However, frequently the deterioration occurs slowly such that it is not expected to interfere with plant operations or safety over plant life. Extensive analytical and experimental efforts by both government and industry have characterized effects of numerous degradation mechanisms that operate in NPPs. These mechanisms vary widely in terms of their potential effects. Some mechanisms are operative in numerous systems, structures, and components (SSCs) over wide ranges of environment and stressor levels; effects of other mechanisms are limited to specific components or materials over narrow ranges of conditions. Degradation mechanisms of concern in NPPs include the following:

Corrosion is a normal phenomenon in NPPs; bulk corrosion in air, water, or steam causes buildup of corrosion products, loss of materials, and activity transport in irradiated systems; however, it seldom results in major thinning or wall penetration; several localized corrosion processes occur in NPPs, such as crevice corrosion, pitting corrosion, galvanic corrosion, various types of stress- or irradiation-enhanced corrosion, etc. These processes can lead to significant local wall penetration and to various, high-consequence failures.

Erosion caused by high velocity steam, water, or two-phase mixtures has contributed to major failures in NPP SSCs. Special processes of importance include erosion/corrosion (local erosion causing a wearing away of a protective oxide film, allowing corrosion to proceed) and cavitation.

Embrittlement of metals and plastics because of structural or chemical changes induced by radiation, high temperature, or atmospheric contaminants can lead to fragility and failure under dynamic loading. Radiation embrittlement and loss of toughness in metals is a consequence of intense neutron radiation fields; thus, only components in close proximity to the reactor core such as pressure vessels and vessel supports are affected. Organic and electronic materials are particularly susceptible to ionizing radiation damage and to degradation caused by attack of aggressive chemical species produced in ionizing radiation fields. For example, ionizing radiation reduces the toughness and dielectric strength of plastic insulators used in cables, motor windings, and transformers. Thermal embrittlement is associated with chemical or metallurgical changes and results from processes such as high temperature thermal aging leading to reduced toughness of ferrous alloys, high temperature sensitization to intergranular stress corrosion cracking in austenitic stainless steels, and oxidation or cross linking of polymers with a resultant loss in toughness and dielectric strength. Hydrogen absorption by ferrous alloys can also lead to loss of toughness and brittle fracture.

Wear is a general concern for rotating or other sliding surfaces where tolerances can affect performance. Lubricant loss or degradation, e.g., because of contaminants or chemical breakdown, can greatly accelerate wear.

Fatigue is a common degradation process that occurs in rotating or vibrating equipment or under other service conditions that place periodic or cyclic loads on a system or component. Associated failures may occur at either high or low cycles in response to various kinds of loads, e.g., vibrational loads, thermal cycles, or pressure transients.

Chemical/biological degradation involves oxygen-induced degradation of organic materials, micro-biologically influenced corrosion (MIC), and plugging and corrosion effects from macro-biological species.

Shrinkage or creep can occur in most materials and are common phenomena in metals at high temperatures. Polymers and composites used as electrical insulators, supports, and protective coatings may exhibit dimensional instability as a result of exposure to high temperatures, moisture, mechanical stresses, or radiation. These effects can degrade insulating and structural properties. Shrinkage of concrete in NPPs is primarily caused by long-term dehydration. Dimensional changes in concrete as it ages do not degrade the properties of concrete; however, when these dimensional changes cause interference, e.g., with other components in prestressed reinforced concrete structures, degradation can occur. Shrinkage is the main contributor to the loss of prestressing forces in prestressed concrete containment.

#### B.2.2 Degradation Sites

Certain sites on or within a given component will normally exhibit more degradation than other sites; and, for many components, degradation is limited to specific locations. Factors that affect vulnerability to degradation include localized chemical or metallurgical variations, geometry with respect to fluid flow or chemical potential gradients, proximity to mechanically or

chemically incompatible materials, and high local stresses. Examples of site-specific degradation include 1) localized erosion/corrosion in ferritic steel piping because of local high fluid velocities, 2) intergranular stress corrosion cracking in heat-affected zones near welds in austenitic stainless steel, 3) excessive hinge pin wear in check valves subject to flutter, 4) rapid degradation of pump impeller blades when cavitation occurs, 5) wear or galling of sliding contacts, 6) crevice corrosion, and 7) fatigue cracking in regions experiencing tensile stresses. Knowing which sites degrade, by what mechanisms, and at what rates is necessary for selecting monitoring methods and for determining where, how, and with what frequency monitoring must be applied to reliably trend and mitigate aging.

### **B.3 MANAGING AGING**

When the mechanisms that cause aging are understood, they can be managed to ensure that aged SSCs will adequately perform their designed safety functions. Inspection, surveillance, condition monitoring, recordkeeping, trending, maintenance, and refurbishment/replacement programs are effective tools for managing aging in operating NPPs. Depending on their intended function, these programs take various forms, such as corrective maintenance, preventive maintenance, predictive maintenance, reliability centered maintenance, condition monitoring, inspections, tests, and surveillance programs. Together, these programs comprise a comprehensive maintenance program that can effectively mitigate potential consequences of age-related degradation.

#### **B.3.1 Inspection, Surveillance, and Monitoring**

Inspection, surveillance, and monitoring methods (ISMM) should reflect mechanistic and empirical assessments performed by qualified staff in their efforts to understand and mitigate aging. These methods which also include testing and condition monitoring should employ state-of-the-art non-destructive examination, such as ultrasonic testing, signature analysis, vibration analysis, dielectric performance measurements, and other measuring techniques performed by qualified or certified staff. Measurement results should be documented, trended, and analyzed with respect to frequency and nature of preventive and corrective maintenance, and for indications of their effect on residual SSCs lifetimes.

The methods of inspection and surveillance, non-destructive examination and condition monitoring are described in more detail as follows:

- **Inspection and Surveillance**

Detailed and comprehensive requirements for monitoring degradation in SSCs are conveyed by the inservice inspection requirements in

Section XI of the ASME Boiler and Pressure Vessel Code and the surveillance testing requirements stated in the plant technical specifications, which are part of the final safety analysis report. These oversight programs can provide useful indications of aging-related deterioration. These programs are supplemented by non-mandatory inspection, surveillance, and testing.

- Non-Destructive Examination

A variety of non-destructive examination (NDE) techniques are employed as part of the inservice inspection and testing programs to detect and characterize flaws or other evidence of degradation that may precede failure. Commonly used methods include visual inspection, dye-penetrant and magnetic particle treatments, radiography, eddy current testing, ultrasonic testing, electrical signature analysis, and acoustic emission monitoring. Each of these methods has advantages and limitations. The limitations derive mainly from the fact that NDE techniques were developed primarily as quality control tools for detecting manufacturing flaws. New or improved NDE methods are continuously being developed. It is expected that techniques that will be available in the future will provide the quantitative characterizations of flaws required for fracture mechanics analysis and will allow on-line monitoring of deterioration in mechanical properties during long-term inservice exposure.

- Condition Monitoring

Condition monitoring and trending of degradation form the basis for predictive maintenance. The overall goal of the predictive maintenance program is to provide information concerning degradation rates and residual lifetimes that can be used to predict and prevent failures. Tools used in doing this include NDE, residual-life assessment, and information analysis and trending. Trends and defined action levels provide guidance needed by the preventive maintenance program to schedule services with a frequency that will avoid failure of critical SSCs. Observed degradation provides opportunities for identifying and eliminating sources of unnecessary deterioration through root cause analysis and corrective action.

### B.3.2 Recordkeeping

Recordkeeping is an essential element of both monitoring and maintenance programs. The product of aging monitoring programs is information that must be translated into effective maintenance practices. This requires 1) that the information obtained by monitoring activities be recorded in adequate, unambiguous detail in a form that is readily retrievable and 2) that the information be easily applicable to specific maintenance practices that effectively mitigate age-related degradation. Records that meet these requirements are needed to prioritize maintenance resources and to correlate actual operating environments and stressors with design assumptions and computed lifetimes so that actual SSCs lifetimes and maintenance intervals can be anticipated.

Maintenance records serve primarily to establish performance histories for the SSCs that comprise the plant. This information and its continuous feedback are useful in specifying what, how, and when equipment should be maintained; what information should be collected; and how it should be recorded. Maintenance histories and equipment performance trends should be documented consistently with regulatory requirements, and the licensees' goals and objectives should be kept current.

A well-developed plant records system, coupled with effective access to experience from similar plants, provides the basis for accurate condition or failure profiling. Effective profiling is important in making proper decisions regarding maintenance actions, that may mitigate the relatively large number (30%) of abnormal occurrences that have been attributed to maintenance shortcomings. Requirements for records retention and retrieval should be established by the maintenance program and be consistent with quality assurance program requirements.

### B.3.3 Trending

Trending of information obtained by monitoring activities can lead directly to maintenance recommendations. Often, however, the selection of the information to be trended must be based on a thorough understanding of the important mechanisms, if the trending results are to be meaningful.

Records of component failure data can be trended and monitored to assess maintenance program effectiveness. Process indicators, such as post-maintenance test results, surveillance test results, ratio of preventive to corrective maintenance, maintenance backlog, and rework frequency, can also be trended to indicate problem areas and overall maintenance effectiveness.

### B.3.4 Maintenance, Refurbishment, and Replacement

Maintenance methods range from simple, straightforward tasks to complex activities that require extensive coordination, training, and technical expertise. The level of oversight and resources devoted to these methods should reflect their complexity and importance to plant safety and reliability. A maintenance program has several important elements. Those relevant to aging management include corrective maintenance, predictive maintenance, preventive maintenance, and refurbishment and replacement. Some of these elements are interdependent and are related to aging monitoring activities discussed in the preceding subsections. In addition, the scope and nature of the various maintenance elements should reflect the as-built plant design specifications; manufacturer's recommendations; operating experience--both internal and external; relevant recommendations and information from the NRC, the nuclear power industry, and its vendors; and general good engineering practices.

- Corrective Maintenance

Corrective maintenance is performed to restore failed or malfunctioning equipment to service. For some types of equipment or components, where failure does not present severe consequences, a

corrective, as opposed to preventive, approach is preferred (e.g., replacement of light bulbs and certain categories of non-critical components). As with other maintenance activities, corrective maintenance priorities should be based on the relative importance of the equipment and on plant safety and reliability objectives. An added function of corrective maintenance is to determine root causes of malfunctions and carry out appropriate corrective action to prevent unnecessary recurrences.

- Predictive Maintenance

Predictive maintenance involves the actions necessary to monitor, find trends, and analyze parameter/property and performance characteristics or signatures associated with a piece of equipment which would indicate that the equipment may be approaching a state such that it may no longer be capable of performing its intended function. The predictive maintenance program should be effective in reducing the failure of SSCs by using techniques that indicate the need for preventive maintenance prior to equipment failure. The data gathered should be analyzed, trends should be identified, and action levels should be defined. Action should be taken to provide feedback to the maintenance program in time to preclude equipment failure. The predictive maintenance program should provide data to the preventive maintenance program and provide and retrieve equipment history data. Root causes should be determined, if possible, and action taken and results fed back into the program.

- Preventive Maintenance

Preventive maintenance includes scheduled actions performed to prevent equipment failure. Malfunctions that represent significant challenges to plant safety or reliability should be prevented. A major responsibility of the maintenance organization is to be cognizant of the significance of potential malfunctions and to ensure that severe consequence events are normally averted by adequate preventive maintenance. Preventive maintenance relies heavily on information generated by monitoring programs to define necessary activities and the frequency at which they should be performed. In addition to input from monitoring programs, preventive maintenance action should be based on equipment histories, other plant performance experience, vendor recommendations, and good engineering practice, including as low as reasonably achievable considerations. Planned actions and schedules should be documented, and departures from these plans should be justified on technical grounds and subject to management review and approval. Clear, comprehensive procedures are vital for preventive maintenance and other oversight and maintenance activities.

- Refurbishment and Replacement

Certain classes of nuclear equipment are subject to refurbishment or

replacement on prescribed schedules. It is important that timely detection of refurbishment and replacement needs that arise in key safety-related equipment due to premature aging are accounted for, both in categories on prescribed schedules and those not on periodic programs.

- Environment Modification

Modifications to water chemistries in BWR reactor coolant systems by hydrogen addition and changes from phosphate to volatile treatments in PWR secondary systems are examples of major measures to decrease corrosion of materials in safety-related areas of NPPs.

#### B.4 DETERMINING POTENTIAL IMPACT OF AGING ON SAFETY

Plant safety could be compromised if degradation of key SSCs is not detected before a loss of functional capability, and timely corrective action is not taken. If unmitigated, age-related degradation of SSCs can result in an undetected reduction in the defense-in-depth concept. The defense-in-depth concept requires that the public be protected from the accidental release of fission products by a series of multiple barriers and engineered safety systems.

A survey (Murphy et al. 1984) of licensee event reports (LERs) conducted by Oak Ridge National Laboratory (ORNL), as part of the planning for this aging research plan, offers examples of numerous instances of aging-induced failures of equipment that have been reported. The reported events indicate that essentially all types of safety-related systems have been affected by a variety of degradation processes. Also, ORNL described the background of selected age-related LERs in detail to provide a better perspective regarding the safety significance of aging (Murphy and Vora 1985). Results of numerous NPAR studies further augment the perspectives of aging impacts. Based on these studies, aging effects can contribute to both the probability of initiation of transients and accidents, and the probability of failure of the mitigating equipment during operation.

The regulatory concern is to recognize and correct or avoid age-related degradation before it decreases the reliability of systems and components to function properly. For example, if maintenance programs are deficient, valves may fail to open or close on demand, sensors may send faulty signals, electrical relays may not operate properly, or diesel generators may fail to start. Individually, such events may not have substantial effects on plant safety; however, the cumulative result may be to decrease the margin of safety. Moreover, failure of a system or component to function properly may subject other components to increased stress and increase the likelihood of their failure.

A possible consequence is that a malfunction of one component could trigger a sequence of component failures, or result in the simultaneous, common-cause failure of degraded components in other systems.



Of particular concern is the potential for common-cause failure of redundant components in safety systems, where simultaneous failure during a transient or accident could lead to loss of functional performance of parallel safety systems. Aging-related degradation that could lead to common cause failures needs to be understood as a basis to preclude accident initiation or loss of safety function and to maintain capability for accident mitigation.

Another type of common-cause failure is the simultaneous or sequential failure of multiple SSCs due to an external trigger event. A classic hypothetical example might be a seismic event that triggers the failure of supports or snubbers restraining corroded piping. Although such occurrences are extremely unlikely, the consequences could be quite severe. The first age-related safety analyses tended to concentrate on this type of major event, which has low probability but potentially severe consequences. In these early investigations, a deterministic approach was used to attempt to calculate the risk associated with plant aging. More recent investigations have used an approach that incorporates the probabilities that various sequences might occur. Both types of analyses include consideration of the likelihood of the sequence or sequences occurring, and both define "risk" to be the product of the probability of occurrence of the sequence and its consequences. The primary difference lies in the methodology used to define which sequence, or sequences, should be considered in the analysis.

#### B.4.1 Deterministic Safety Analysis

In a deterministic analysis, the choice of sequence and the sequence probability is developed through the technical judgments of an interdisciplinary panel of experts. An example would be the Appendix K analyses of effectiveness of emergency core cooling systems (10 CFR 50). As other examples of these kinds of judgments, we have 1) the choice of a "guillotine" pipe break for large loss-of-coolant analyses, 2) the more recent acceptance of the "leak before break" criterion, 3) the complicated rules detailing which mitigating equipment is available after the initiating event, and 4) the single failure criteria. The types of decisions and choices selected are typically based on an intuitive understanding of which accident sequences are likely to justify analysis, that is, to be significant contributors to the overall risk associated with the plant.

#### B.4.2 Probabilistic Risk Assessment

In contrast to the deterministic approach, where the choices of sequence are based on expert judgment, the probabilistic risk assessment (PRA) approach attempts to use the likelihood of occurrence and the consequences associated with the occurrence to determine which sequences to analyze, and their probable contribution to overall plant risk. The methodology used in a PRA focuses on the risk associated with each potential sequence to determine which sequences justify analysis. In addition, PRAs normally include fault tree analyses, which provide enumeration of the specific sets of component (and other basic) failures that can cause a particular sequence of events leading to an accident.

In practice, because of limitations in the availability of precise failure rate data and the impracticality of performing extensive Monte Carlo type calculations involving all plant SSCs, expert judgment is used to constrain the types of sequences considered in a PRA. Likewise, the expert judgment that is used to select sequences to be considered in a deterministic safety analysis is based on a knowledge of historical failure rates and an implied perception of risk. Thus, it can be argued that both "deterministic" and "probabilistic" safety analyses are hybrids, each encompassing elements of the other's methodology, differing only in the time and manner in which time elements enter into the overall process.

#### B.4.3 "Hybrid" Analysis

The term "hybrid" has a special meaning in regard to safety analyses for license renewal. Specifically, the hybrid safety analysis approach uses the deterministic safety analysis methodology to quantify risk associated with age-related degradation of vital components, where the most significant accident sequences are well defined. The PRA methodology is primarily used to screen nonvital systems and components to determine if failure sequences within them could adversely affect the performance of vital systems and components, thus increasing overall plant risk. This combination, or hybrid, approach takes advantage of the lower costs associated with the deterministic safety analysis, while assuring against unforeseen accident sequences that might not have been recognized without application of the probabilistic safety analysis methodology.

#### B.5 HUMAN-MACHINE INTERFACE

Human factors have contributed to accidents and are recognized as a major factor in nuclear plant safety. The effects of personnel on the aging of nuclear equipment have been identified in Phase-I assessments conducted under the NPAR Program. For example, the diesel generator aging study recognized classes of failures in the historical database that were attributed to inappropriate personnel practices.

Human factors relate to plant aging in two ways. First, they can contribute to the aging of plant systems and components by affecting materials, stressors, environments and their interactions. For example, inferior maintenance of diesels will contribute to their aging, or faulty application of procedures during inservice inspection will fail to detect material flaws. Second, human factors are important in the response to aging phenomena. For example, by understanding how aging affects plant performance, plant personnel are better prepared to make decisions relating to the operation and maintenance of a plant. Also a plant staff that is alert and perceptive to unusual phenomena become a defense against the consequences of aging.

APPENDIX C

ONGOING PROGRAMS RELATED TO NPAR

## APPENDIX C

### ONGOING PROGRAMS RELATED TO NPAR

#### C.1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) and regulatory agencies in foreign countries are sponsoring research and development programs to evaluate the impact of aging on the safe operation of nuclear power plants (NPPs). The nuclear industry, including the Electric Power Research Institute (EPRI), nuclear steam supply system (NSSS) vendors, utilities, and architect engineers, is interested in aging because of the significant economic advantage that can be gained from extending the operating license of aged power plants. Therefore, the nuclear industry, EPRI, and the U.S. Department of Energy (DOE) are sponsoring pilot studies to evaluate the issues involved in renewing operating licenses for BWR and PWR plants.

Several NRC and industry-sponsored programs related to aging of NPPs are currently underway. Many of these programs are complementary to each other and, therefore, it is essential that they are coordinated. Coordinating these programs eliminates duplication of effort and provides more complete coverage of aging-related issues in a timely and cost-efficient manner. Such coordination and integration has a special significance for NRC if the criteria for license renewal (LR) are to be formally established in the near future.

The Executive Director for Operations established a Technical Integration Review Group for Aging and Life Extension (TIRGALEX) to ensure effective utilization of NRC resources. TIRGALEX recognized the importance of the coordination effort and conducted a preliminary review of relevant programs and activities already under way. Further coordination across NRC offices was provided by the Aging and Life Extension Coordinating Committee (ALEXCC). Currently, the NPAR Review Group, comprised of representatives from the NRC offices and regions, provides review and coordination for age-related research programs.

Interfaces between the NPAR Program and other ongoing NRC programs have been established and will be maintained. Similarly, external programs involving both domestic and foreign organizations have been contacted. This section outlines the aging-related programs sponsored by NRC. It also outlines the similar programs sponsored by industry in the United States and foreign countries. The outline of these programs is structured into the following subsections:

C.2 NRC Programs and Activities Related to NPAR

C.3 Programs Jointly Sponsored by industry, EPRI, and DOE

#### C.4 Other EPRI Programs Related to NPAR

#### C.5 Ongoing Aging and Life Extension Programs at DOE

#### C.6 Ongoing Life Extension Activities in Codes and Standards Committees

#### C.7 Aging and Life Extension Programs in Foreign Countries.

The list of age-related programs external to NRC will be updated periodically. The updating will be made as information exchanges take place and information for new programs is made available to the NPAR Program.

### C.2 NRC PROGRAMS AND ACTIVITIES RELATED TO NPAR

In this section, a summary of ongoing NRC aging and LR related programs and activities is presented.

#### C.2.1 Office for Analysis and Evaluation of Operational Data (AEOD)

The AEOD collects, screens, evaluates, and reports on operational experience and data from the entire nuclear industry. Through this process the AEOD identifies components and systems with repetitive, unusual, or possibly generic failure modes, some of which are aging related.

The AEOD also conducts evaluations of components, systems, procedures, and management structures in response to specific operational events. In performing these evaluations, AEOD is concerned with acquiring and analyzing data that lead to identifying the root cause(s) of the events. The AEOD does not specifically focus on aging issues per se, unless the incident involves failures or degradations due to aging mechanisms. When an event indicates age-related degradation, the information is forwarded to the appropriate NRC office for action. The AEOD then publishes case studies that contain detailed descriptions of the failure parameters, failure causes, and engineering evaluations.

The AEOD has developed, through the Oak Ridge National Laboratory (ORNL), a comprehensive computerized data base to aid in collecting and evaluating licensee event reports. This data base, the Sequence Coding and Search System (SCSS), contains established procedures and codes for collecting operational data. The SCSS also provides a means of storing and rapidly retrieving root-failure-cause data that can be used in studies of component life and component aging. It also has resulted in codes specifically identifying age-related degradation.

The AEOD has an ongoing project to analyze the Nuclear Plant Reliability Data System (NPRDS) data base. Jointly with the Institute of Nuclear Power Operations, which maintains the data base, AEOD has developed a list of critical components on which to focus attention. The trends and patterns analysis of NPRDS data focuses on these key components. The AEOD's analysis of NPRDS data results in statistical and engineering evaluations of component

failure modes, times to failure, operating conditions that affect failure, and chemical and physical conditions affecting component-wearout rates.

### C.2.2 Office of Nuclear Reactor Regulation (NRR)

A substantial number of aging-related programs are in progress in NRR. These programs are discussed below in terms of Program categories, Regional activities, and Initiatives.

#### Program Categories

The NRR programs fall under three broad program categories: Operating Reactors, Casework, and Safety Technology.

The Operating Reactors categories includes licensing actions and safety assessments of both currently operating reactors and new reactors coming on line. Although aging concerns are generally addressed in the licensing process, to some extent, the licensing process does not consider the aging processes for all SSCs. The capability of equipment to perform satisfactorily for its specified lifetime is NRR's principal area of attention.

Casework is one of the larger program categories in NRR and represents projects that are conducted to support individual licensing actions. These projects may be initiated by licensee requests for license amendments or by events that occur at the plants that may require a regulatory response from the NRC. Most aging-related projects are included in this program category. NPAR expertise has been utilized in certain case work areas, e.g. service water systems.

The Safety Technology program category is divided into five subgroups: unresolved and generic safety issues; risk assessment; regulatory requirements; code analysis and maintenance; and human factors program issues. Some of these programs have aging-related aspects. For example, unresolved and generic safety issues are sometimes related to aging issues, e.g., Pressurized Thermal Shock. In the past probabilistic risk assessments (PRAs) have not considered all potential effects of age-related degradation; however, NPAR is developing methods to incorporate effects of aging into PRAs.

Within the human factors program area, the crucial role of maintenance in predicting and correcting age-related degradation has been reflected in the Maintenance and Surveillance Program. Phase I of the Maintenance and Surveillance Program (NUREG-1212, Vol 1 & 2), was designed to survey current maintenance practices in the U.S. nuclear industry and to evaluate their effectiveness. Phase II is working toward resolutions of the issues identified and assessed in Phase I.

#### Regional Activities

Several NRR programs guide ongoing regional inspection activities relevant to aging, aging detection, and mitigation of aging consequences.

These programs include the Safety System Functional Inspection Program, the Safety System Outage Modifications Inspection Program, and the Generic Communication Program.

The Safety System Functional Inspection Program, in general, assesses whether plant modifications of selected safety systems have degraded the design margin to the point where the system's ability to mitigate design basis events is impaired. This program consists of an in depth review of a small number of safety systems and is usually conducted at older plants. The major objectives of the program are to ensure that:

- testing is adequate to demonstrate that the safety systems are capable of performing the safety functions required by their design bases
- system maintenance (with emphasis on pumps and valves) is adequate to ensure system operability under postulated accident conditions
- operator and maintenance technician training is adequate to ensure proper operation and maintenance of the systems
- human factors considerations relating to systems and supporting procedures are adequate to ensure proper system operation under normal and accident conditions.

The objectives of the Safety System Outage Modifications Inspection Program are to verify, through sampling inspections, that:

- licensees have instituted effective controls for conducting modification and repair activities during outages
- activities are accomplished in accordance with established procedures and commitments
- completed repairs and modifications have been properly designed, installed, inspected, and tested
- affected systems are ready for safe startup and operation of the plant following an outage.

The objectives of the Generic Communication Program are to:

- inform licensees of problems, including those due to aging and wear, that have developed in individual plants
- appraise licensees of required action when these problems are shown to be significant and generic.

These three programs apply to the pressure boundary hardware, drivers and actuators, electrical power, and the instrumentation and controls of engineered safety features.

The NRR also guides the activities of the regions by issuing the Inspection and Enforcement Manual. Portions of this manual establish inspection procedures that are relevant to aging and life extension. For example, some inspection procedures establish guidance for ascertaining that inservice inspection and testing activities are programmed, planned, conducted, recorded, and reported in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (ASME 1986). Where applicable, these procedures prescribe inspection of the licensee's recordkeeping for modification, maintenance, and repair activities.

The components selected for specific inspection efforts are containment, instruments and components (specifically, breakers and relays), and reactor coolant system piping. The programs in the component area are directed at breakers and relays that may cause anticipated transients without scrams and/or loss of safety-related equipment. The containment programs address corrosion, fatigue, and crack propagation in steel, reinforced-concrete, and prestressed-concrete containments. The reactor coolant system piping program addresses stress-corrosion cracking of welds, and erosion and general corrosion of pipe bends and elbows.

### NRR Initiatives

NRR has undertaken two major initiatives in preparation for the planned receipt of LR applications for the lead nuclear power plants (Yankee Rowe in June 1991 and Monticello in December 1991). The first major initiative involves participation with RES in development of the LR Rule and the supporting rulemaking package on LR. The Rule (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," is scheduled to be published in June 1991. The second major initiative involves the development of a Standard Review Plan for License Renewal (SRP-LR) which the staff will use in the review of LR applications. Also in support of the Rule, the Office of Regulatory Research has issued a draft regulatory guide (DG-1009), Standard Format and Content of Technical Information for Applications to Review Nuclear Power Plant Operating Licenses. Both the draft SRP-LR (NUREG-1299) and the draft regulatory guide were issued for comment in December 1990; publication of the final documents is scheduled for April 1992.

### C.2.3 Office of Nuclear Regulatory Research (RES)

RES sponsors a number of large and important aging-related programs. The general objectives of these programs are to identify aging mechanisms, evaluate their safety and regulatory impacts, assess detection methods for particular aging degradation mechanisms, and develop mitigating and corrective actions.

Major program areas with aging-related projects include Mechanical and Electrical Equipment Qualification; Instrument Integrity; Primary System Integrity, which includes the Heavy Section Steel Technology program, Degraded Piping, and Steam Generator Integrity projects; Equipment Operation and Integrity, which includes the Nuclear Plant Aging Research Program; and Non-destructive Examination.



RES has an initiative underway addressing 10 CFR Part 54 to develop and publish (jointly with NRR) a License Renewal Rule. The Rule is scheduled for publication in June 1991. Another RES activity addresses 10 CFR Part 51, to develop and publish an environmental Rule for LR. The proposed Rule is scheduled for publication in August 1991, and the final Rule is scheduled for July 1992.

### C.3 PROGRAMS JOINTLY SPONSORED BY INDUSTRY, EPRI, AND DOE

A joint industry, EPRI, and DOE program was initiated in 1984 to identify issues associated with light-water reactor (LWR) license renewal. In 1984, DOE and EPRI agreed to co-fund studies and developed a joint R&D plan in 1985. Two pilot studies involving Surry-1 (PWR) and Monticello (BWR) were initiated. After completion of the pilot studies, a bidding process was used to select two lead plants to develop requests for LR for submittal to NRC.

Northern States Power conducted the BWR pilot plant study at its Monticello Nuclear Generating Station, co-funded by EPRI and DOE. The utility was the successful bidder for developing a lead plant LR request, also for Monticello. The request is expected to be submitted to NRC in December 1991.

The PWR pilot plant study was conducted by Virginia Power at its Surry Unit-1 plant. Yankee Atomic Electric was the successful bidder for developing the PWR lead-plant LR application, for the Yankee Rowe Nuclear Power Station. The LR application is scheduled to be filed with the NRC in September 1991.

In June 1988, the Nuclear Management and Resources Council, composed of representatives from all U.S. nuclear utilities, formed its NUPLEX (Nuclear Utility Plant Life Extension) Working Group, and charged NUPLEX with the lead responsibility for all nuclear industry activities related to LR. The collaborative efforts of NUPLEX, EPRI, and DOE have resulted in the issuance of eleven industry reports (IRs) in support of the LR process. These IRs, which have been submitted to the NRC for review (see Figure 5.2), are intended to provide the generic technical bases for evaluating the effects of age-related degradation in specific NPP structures and systems important to LR.

### C.4 OTHER EPRI PROGRAMS RELATED TO NPAR

In addition to the joint programs noted in section C.3, EPRI has a number of programs related to plant life cycle management that address age-related degradation. Examples include:

1. Nuclear Plant Life Extension
2. Corrosion Control
3. Component Reliability
4. Steam Generator Reliability.

The Nuclear Plant Life Extension Program approaches aging and life extension from the technological perspective of understanding the aging process for systems and components. It does not address institutional or legal issues. The Corrosion Control Program addresses environmentally caused cracking and pitting. It specifically emphasizes BWR water chemistry and understanding and mitigating pipe cracking due to corrosion. The Component Reliability Program is related to the Corrosion Control Program. Structural reliability and safety improvement are the main thrusts of the Component Reliability Program. Specific examples of activities include material characterization, flaw detection and assessment, inspection hardware development, and reliability methodology. While the central program focus is materials, the program also addresses piping, reactor pressure vessel, and steam generator tubes.

Three other EPRI initiatives, are of special interest to the NPAR Program; these are the NCIG (formerly, the Nuclear Construction Issues Group), the Nondestructive Evaluation (NDE) Center, and the Nuclear Maintenance Applications Center (NMAC). The primary activity of NCIG of interest to NPAR is the preparation of guidelines for recordkeeping related to procurement, qualification, and replacement of nuclear components. The NDE center has compiled an impressive record in transforming advanced NDE methods from promising techniques into proven, field-qualified inspection technologies. The NMAC develops guidelines for maintenance and maintenance-related activities (e.g., pre- and post-maintenance equipment monitoring and diagnostic techniques).

EPRI LR-related activities in CY 1991 include:

- participation in lead plant initiatives (~50% of funds)
- participation in industry technical reports (~30% of funds) including resolution of review comments
- data and record guidelines
- cable condition monitoring
- in-plant versus artificial aging.

#### C.5 ONGOING AGING AND LIFE EXTENSION PROGRAMS AT DOE

Plant license renewal is being pursued by DOE in several ways. One of three major efforts within the LWR safety area is the cooperative industry/EPRI/DOE program to extend the productivity of existing and future LWRs. DOE has conducted a study on delineating the key LWR structures and components capable of continued service beyond their design life without refurbishment or replacement. Further, DOE has evaluated items requiring replacement in terms of safety significance, licensing, and cost implications. The DOE programs are intended to support relicensing by influencing the federal regulatory process and evaluating the safety and economic impacts of improved plant performance.

Sandia National Laboratory (SNL) is the lead DOE laboratory for research related to LR. Principal activities sponsored by DOE/SNL in FY 1991 are:

- participation in lead plant initiatives (Yankee Rowe and Monticello), including identification of effective programs to manage aging
- participation in preparing 10 industry technical reports (first in August 1989) (issued October 1989 through October 1990), including resolution of review comments
- emergency diesel laboratory at SNL
- development of screening methods
- evaluation of flaw distributions in reactor pressure vessels
- evaluation of sub-sized Charpy specimen performance
- vessel support embrittlement studies
- development (with GE) of in-core electrochemical potential probes
- development (with GE) of an in-reactor crack monitor
- distribution of cable aging report
- annealing studies on reactor pressure vessels.

#### C.6 ONGOING LICENSE RENEWAL ACTIVITIES IN CODES AND STANDARDS

The following is a summary of the status of code activities related to license renewal:

Institute of Electrical and Electronics Engineers (IEEE). Working Group 3.4, "Nuclear Plant Life Assessment," is chartered to investigate the codes and standards aspects of plant license renewal as they pertain to Class IE electrical equipment. The working group has developed an action plan and has provided several drafts of a guide document to function in place of standards.

Instrument Society of America (ISA). There are no current activities. They take their lead from IEEE. When relevant items arise, IEEE will make the necessary contacts.

American Society of Mechanical Engineers (ASME). Under the auspices of the ASME Section XI subcommittee, the Special Working Group on Plant Life Extension is active. They have participation from EPRI, NRC, DOE, utilities, several laboratories and NSSS suppliers. To date, the special working group has reviewed a broad spectrum of aging and license renewal programs.

Several important changes and additions were incorporated in Section XI of the ASME Boiler and Pressure Vessel Code during 1988 and 1989.

### C.7 AGING AND LIFE EXTENSION PROGRAMS IN FOREIGN COUNTRIES

Extensive programs in materials degradation, NDE, fracture mechanics, structures, and other aging-related areas exist worldwide. A few of these have been detailed in NUREG/CR-3040. The NRC operates large cooperative international research programs, several of which are directly related to aging and life extension.

Other programs identified in the analyses of several selected components and structures include:

- Westinghouse and Framatome--performing work separately and jointly on NDE for cast stainless steel.
- JAERI--evaluating the effects of separate and simultaneous application of environmental stressors on cables (NUREG/CP-0071).
- France--developing predictive maintenance programs.

Additional details on foreign activities on NPP aging and life extension are summarized below.

#### CANADA

The AECL (Atomic Energy of Canada Limited) approach to managing aging degradation of nuclear power plant components and maintaining adequate plant safety embodied in their Local In-Plant Fatigue Evaluation (LIFE) System. The LIFE system involves the recording, assessment, and evaluation of the history of the in-plant conditions. Particular emphasis is placed on fatigue evaluation, rehabilitation and replacement of key CANDU reactor and heat transport system components.

As applied to currently operating CANDU (Canadian Deuterium Uranium) reactors, major features of the LIFE approach include:

- Operating policies and procedures prescribe practices that are designed to minimize aging effects. Components, particularly pressure-retaining components, important to safety are subject to inservice inspection.
- Deficiencies are systematically detected and recorded; causes (including aging degradation) are determined, and appropriate corrective actions are taken.

- Licensees are required to monitor and periodically evaluate equipment and system performance against the reliability targets set by the Atomic Energy Control Board (AECB), allowing detection of aging trends.
- Research and development, by both the industry and the AECB, provide the basis for predicting the behavior of critical plant components during plant operation, developing better components and developing monitoring methods capable of detecting component degradation before loss of safety function occurs.
- Aggressive monitoring is utilized, employing monitoring instruments, periodic testing, inspection, maintenance, and field patrols. It is recognized that age-related degradation can be detected only if appropriate time-variant parameters are monitored.
- Significant Event Reports are prepared for events that have a significant negative effect on reactor safety, worker or public safety, and cost. This system has the following important features:
  - Recording of equipment and operation deficiencies in a specified, systematic manner to allow event review and analysis.
  - Multilevel diverse screening process.
  - Trend lines of equipment and system deficiencies.
  - Communication of lessons learned to other CANDU stations, owners, designers, and equipment manufacturers.
- Reactor Safety Reliability Assessment is an important part of the annual comprehensive and systematic review of nuclear power plant operation and maintenance by both the licensee and the AECB. It gives the actual past-year performance and the predicted future performance in terms of system unavailabilities and serious process failure occurrences. These can be compared to the AECB reliability targets mentioned earlier.

Information obtained from the LIFE program is also incorporated in new (CANDU-3) plants during the initial design stages. Special attention is paid to maximize the longevity of the components and where required make provisions for economic and expedient replacement procedures.

Ontario Hydro (OH) also has an internal Nuclear Plant Life Assurance (NPLA) program aimed at:

- maintaining the long-term reliability, availability, and safety of OH's nuclear plants during the normal service life of 40 years (life assurance).

- preserving the option of extending the life of OH's nuclear plants beyond the currently assumed service life of 40 years (life extension).
- ensuring that future activities in these areas in OH will conform to an overall plan.

The NPLA program which is quite similar to that of NPAR, includes initiatives a) retrieve samples and information from the Nuclear Power Demonstration (NPD) reactor, and b) inspect and assess the condition of critical components of the Pickering Station reactors during retubing operations.

### GERMANY

The Gesellschaft für Reaktorsicherheit, m.b.H. (GRS) has been engaged, on behalf of the Federal Minister of the Environment, Nature Conservation, and Regulatory Safety, in a data collection program at power plants for 15 years (since initial operations). This program has demonstrated how long-term data collection and trending of component performance parameters and functional indicators (a strategy/approach similar to that recommended in the NRC's program plan for Nuclear Plant Aging Research NUREG-1144) provide protection against failures resulting from age-related degradation.

Age-related degradation has been observed in emergency diesels; power, measuring, and control cables; pump motors; valves; and a variety of electric and electronic equipment. The safety and availability problems that may result from aging are counteracted by a system of inspection and planned preventive maintenance measures. The GRS has implemented a long-term systematic collection of data so that the accumulation of defects on certain components can be recognized.

### FRANCE

From the outset of the nuclear program, the French Safety Authorities and the Électricité de France (EdF) (the main licensee) took into consideration the effects of aging on the installed equipment.

In French designs, qualification is one means used to check equipment design. In most cases, it includes testing designed to evaluate the behavior of the equipment with time.

Examples of aging experiences in French reactors include:

- For mechanical equipment--diesel generators are exposed to untimely starting; steam generators are subject to unforeseen corrosion or erosion due to foreign matter.

- For electrical equipment--isolating switches are operated under unscheduled loading conditions and batteries whose autonomy sometimes changes unexpectedly.

In addition, materials such as coatings (paints) or lubricants (oils or greases) have a great effect on the behavior of the equipment with which they are associated.

In conjunction with the safety authorities, EdF has initiated a program of investigations to:

- develop a method of measurement, in theory and practice, of the aging of equipment
- determine the influence of NPP operating procedures on aging.

In addition, a file of events is kept up to date for each plant and is analyzed to evaluate the behavior of the installation.

Since the beginning of the French nuclear program, means have been set up that ensure the periodic monitoring of operation or of the intrinsic properties of the equipment or systems. Some typical examples are:

- periodic testing
- inservice inspection
- preventive maintenance.

Finally, procedures have been set up to evaluate primary circuit performance for each plant (measurement of primary circuit pressure and temperature values above a certain threshold) and allow comparison to design features.

## JAPAN

Technology development for NPP life extension is a priority effort in Japan. A 7-year technology development plan was implemented in FY 1985.

The major tasks in plant life extension technology development are related to the diagnosis of NPP aging deterioration, prediction of remaining plant life, and replacement and improvement of plant equipment. The following tasks have been implemented for comprehensive evaluations:

- development of life diagnosis and prediction methodology (developing life evaluation methods, creating a data base on aging and degradation phenomena, developing monitoring techniques)
- development of technology for replacing and improving large equipment (developing methodology, verification).

The selection of equipment and structures to be studied in detail was made in 1985. Structures and equipment that could be easily replaced were not subject to further evaluation. The remaining structures and components were ranked according to evaluation criteria that included ease of replacement and impact on safety.

Research on age-related degradation of key components (e.g., RPV, concrete structures, and cables) is being conducted by the Nuclear Plant Aging Research Team (NPART) at the Japan Atomic Energy Research Institute. Studies on irradiation embrittlement of RPV steels encompass the effects of water chemistry and temperature on crack growth, flux intensity effects, and destructive examination of samples cut from the pressure vessel of the Japan Power Demonstration Reactor (JPDR). Research on age-related degradation of concrete structures includes the development of monitoring methods, testing of samples bored from the biological shield of the JPDR, and separation of neutron- and thermal-aging effects on concrete strength. The cable research program includes development of monitoring methods, accelerated aging of cables, and correlations between aging of test specimens and actual NPP components.

## TAIWAN

A Component Life Assessment for NPP (CLANPP) program in Taiwan involves staff from the Materials Research Laboratories (MRL) and from Taiwan Power Company (Taipower). The program began in July 1987. The project goals are to establish a data base on material properties, failure histories, and operational parameters for selected critical components; also to provide component life prediction criteria.

Current research activities conducted under the CLANPP Program include:

- stress corrosion cracking (SCC) and irradiation-assisted stress corrosion cracking (IASCC) studies
  - SCC: Establishment of SCC crack growth rate prediction model and development of on-line monitoring techniques
  - IASCC: Participation in the tests to study IASCC
- Life assessment of reactor vessel and its internal components
  - Evaluation of fatigue usage factor
  - Radiation induced degradation of support skirt
  - IGSCC/IASCC of Top Guide/Shroud/Core Plate



- Aging study of safety-related valves
  - Motor current monitoring of aged motor operated valves
  - Aging study of solenoid operated valves
  - Improvement of stem packing performance

In addition to participation in the CLANPP Program, NRL has or is planning research activities in the following areas:

- Life assessment on turbine rotors
- Pipe integrity study of LWRs
- Component degradation study of PWR feedwater system
- Aging assessment of cables.

### SPAIN

A plant life extension program is underway in Spain, emphasizing surveillance. The program deals with programming and reporting inspections and tests in the following areas:

- single- and double-phase erosion corrosion
- low-pressure turbine disc cracking
- intergranular stress corrosion cracking of piping, and
- control rod drive housing welds.

Fracture mechanisms and procedures for evaluating remaining life are addressed.

### IAEA Activities

Since 1986 the IAEA has been developing initiatives in nuclear plant aging and life extension. Highlights of the activities are outlined below.

1. In 1986 an IAEA technical committee meeting of "Safety Aspects of Nuclear Power Plant Aging," was held in Vienna, Austria, September 1-5. Delegates from 12 countries attended.
2. In 1987 the IAEA sponsored an international symposium on "Safety Aspects of the Aging and Maintenance of Nuclear Power Plants," held in Vienna, Austria, June 29-July 3. There were 140 attendees from about 30 countries.

3. In 1988, J. Pachner, representing IAEA, outlined IAEA plans for a comprehensive program on nuclear plant aging and life extension at the International Symposium on Nuclear Power Plant Aging, Bethesda, August 30, 31, September 1.
4. In 1989 and 1990, IAEA sponsored the following activities in NPP Aging:
  - June 1989, consultants meeting on NPP recordkeeping, held in Vienna, attended by representatives from four countries.
  - November 13-17, 1989, consultants meeting in "Safety Aspects of NPP Aging," held in Vienna.
  - November 20-24, 1989, technical committee meeting on "The Evaluation and Management of the Safety Impact of NPP Aging," held in Vienna, attended by representatives from 16 countries.
  - June 23-28, 1990, consultants meeting to finalize the technical report on, "Selection of NPP components for Ageing<sup>(a)</sup> Management Pilot Studies."
  - November 5-9, 1990, technical committee meeting to define existing knowledge and gaps relating to understanding and managing of aging and to propose pilot studies, on selected components. Representatives from 14 countries and the committee of European communities attended.

IAEA working groups have been organized to conduct pilot studies on four safety significant NPP components, selected on the basis of their safety significance and their susceptibility to different types of ageing degradation. These components are: 1) the primary nozzle of a reactor pressure vessel, 2) a motor operated isolation valve, 3) the concrete containment building, and 4) instrumentation and control cabling within the containment.

The objectives of the pilot studies are, for each component, to identify dominant ageing mechanisms, to identify or develop an effective strategy for managing ageing effects caused by the identified mechanisms, and to validate or improve the methodology for ageing management studies developed by the IAEA.

A document summarizing the pilot studies is expected to be published in 1991. Implementation of recommendations from the studies will proceed in 1992.

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(a) IAEA spelling.

The basis for cooperation between IAEA and NEA/OECD has been established. Joint participation in a meeting in 1991 has been planned. The duration of the IAEA program in Safety Impact of NPP Ageing is expected to be approximately five years.

The results of the pilot studies will have application in monitoring the degradation and in preventive maintenance of the selected components, including the development of criteria for decisions on the type and timing of preventive maintenance actions; and in predictions of component performance and remaining service life under all expected service conditions, including postulated accident and post-accident conditions.

APPENDIX D

ALTERNATIVE APPROACHES FOR CONDUCTING NPAR  
PHASE-II COMPREHENSIVE ASSESSMENTS

## APPENDIX D

### ALTERNATIVE APPROACHES FOR CONDUCTING NPAR PHASE-II COMPREHENSIVE ASSESSMENTS

Alternative approaches have been used to conduct comprehensive aging assessments of naturally-aged components and systems under the NPAR Program. These approaches are determined by the size and complexity of the components and systems, relating to the equipment availability and potential expense of the aging assessment. The first approach, in-depth laboratory investigation, focuses on laboratory assessments of age-related degradation of components, such as small electric motors. The second and third approaches, systematic applications of experts and in situ evaluations, focus on age-related degradation of systems and components that are too large, complex, or costly for laboratory study. Examples of these components include emergency diesel generators, snubbers, and service water system components. These alternative approaches to comprehensive aging assessments are described below in more detail.

#### D.1 APPROACH I - EXAMPLES OF IN-DEPTH LABORATORY INVESTIGATIONS

##### Case 1 - Small Electric Motors

The small motors are generally sufficiently compact, portable, and available to accommodate laboratory investigations within the available budget. Motors representing a range of sizes and models were investigated at BNL, including seismic tests. Phase-II results are reported by Subudhi, Burns, and Taylor (1985).

##### Case 2 - Motor-Operated Valves

Suitable valves were also available and portable, permitting extensive laboratory studies at ORNL for both aging characteristics and monitoring methods. Phase-II results are reported by Crowley and Eissenberg (1986).

#### D.2 APPROACH II - EXAMPLES OF SYSTEMATIC APPLICATION OF EXPERTS AND IN-PLANT ASSESSMENTS

##### Case 1 - Aging Assessment of Emergency Diesel Generators (EDGs)

The EDGs represent a class of equipment that is large and complex, having approximately 25 models and nine suppliers. Nuclear diesels in the size range at current plants have either not been retired or have been diverted to other uses. Even given one or more naturally-aged EDGs, an in-depth laboratory study (at considerable expense) would address aging characteristics of one

model that may have limited application to other models. The alternative approach selected by PNL investigators (Figure D.1) was to establish a panel of three EDG experts, collectively representing ~80 years of experience with nuclear and non-nuclear EDGs, including experience with design, operation and failure analysis. The panel focused on a statistically selected set of ~2000 EDG failures described in four databases. The panel was directed to systematically interpret the failures (generally minimal) in terms of their root cause and whether their cause was age-related. The panel's interpretation, of the aging characteristics of the nine EDG types were entered into a dBASE III program that accommodated numerous correlations. The results facilitated identification of the EDG components most vulnerable to aging and the prominent aging mechanisms. The results were then subjected to review in a workshop that included EDG operators from several utilities. Results are reported in NUREG/CR-4590, Vols. 1 and 2 (Hoopingarner et al. 1987; Hoopingarner and Vause 1987). A second workshop, involving the EDG experts established the basis for recommendations to mitigate aging of EDGs. Information collected during site visits and observations during a major EDG overhaul validated the

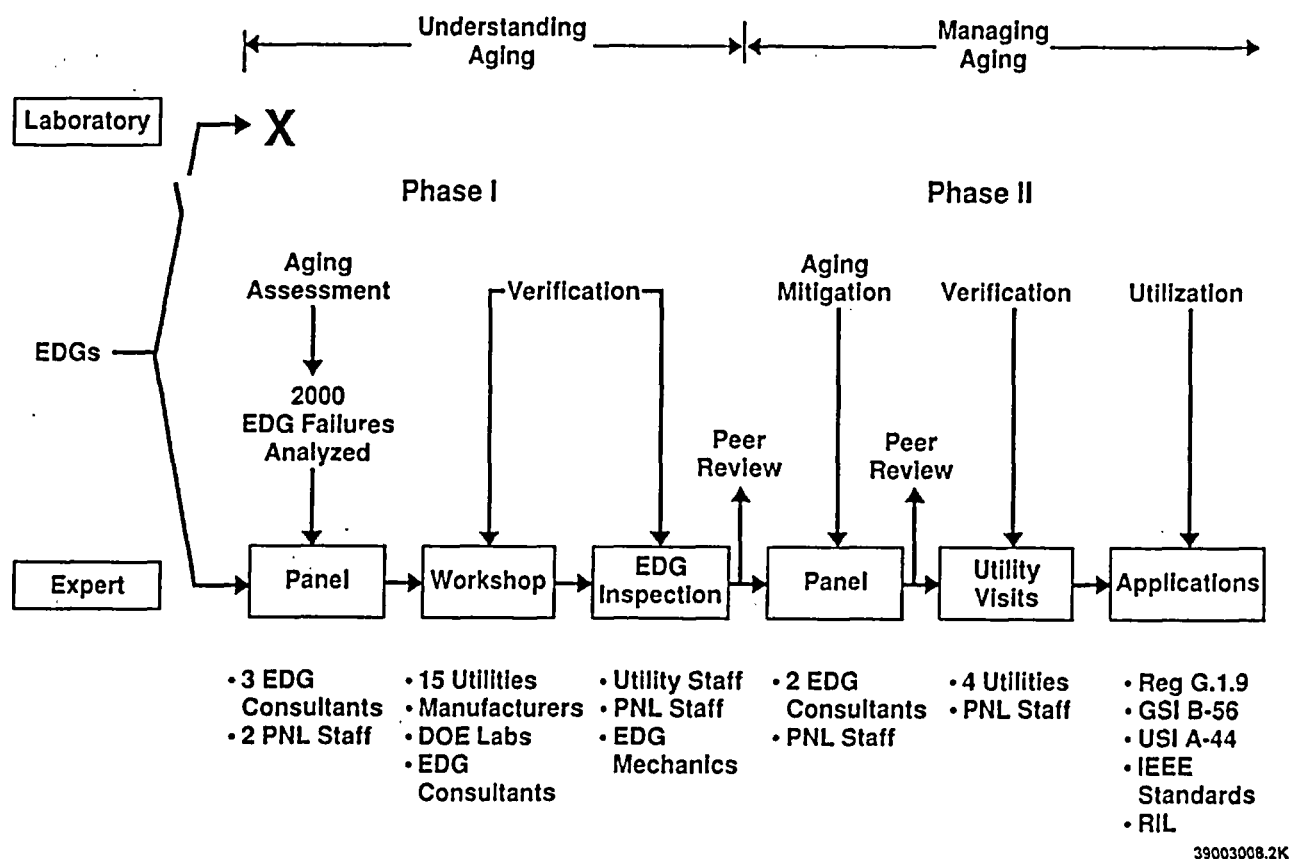


FIGURE D.1. Application of NPAR Strategy to Understand and Manage Aging of EDGs

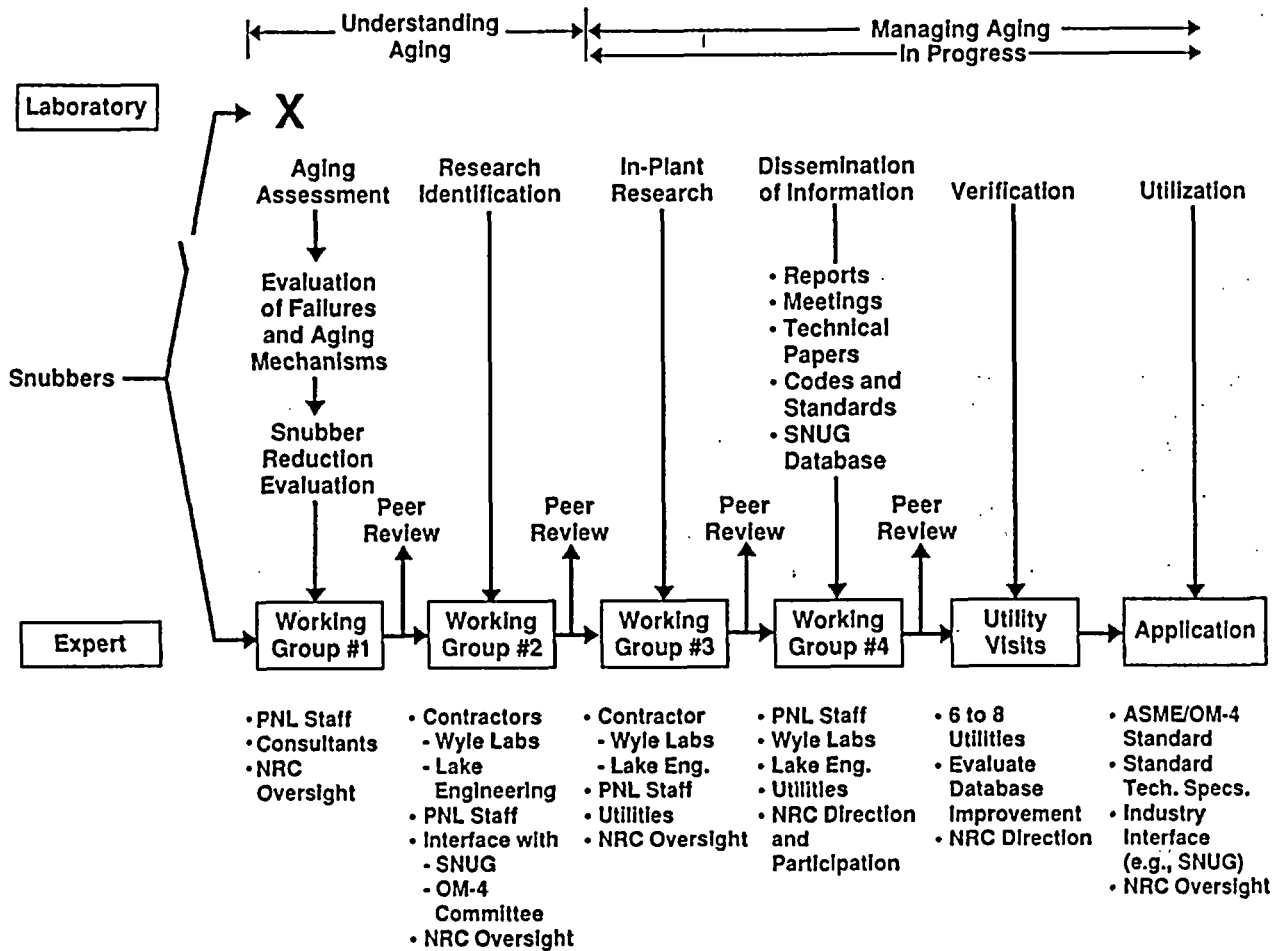
research results. Several regulatory applications resulted from the EDG aging research. The recommended approach to aging mitigation is reported in NUREG/CR-5057 (Hoopingarner and Zaloudek 1989).

### Case 2 - Aging Assessment of Snubbers

Snubbers are components that involve two prominent designs (hydraulic and mechanical) and a wide range of sizes, and applications. A large number (>1,000) are used in many plants, and they are exposed to a wide range of environments. Laboratory studies were feasible but it would be costly to address statistically-significant tests of various snubber sizes and designs under simulated plant conditions. Snubbers occasionally become contaminated with radioactive material during service, thereby increasing examination costs and requiring difficult laboratory procedures and disposal. Therefore, the Phase-II comprehensive snubber aging assessment was pursued using an in-plant snubber aging study (Figure D.2). The in-plant approach involved plant visits to cooperating utilities by snubber experts. The utilities provided access to data from a relatively large number and variety of snubbers. Experts familiar with snubber designs, operating conditions, and failure characteristics participated in acquisition and interpretation of the data.

### Case 3 - Aging Assessment of Major LWR Components and Structures

The major LWR components and structures are designed and fabricated by several vendors and architect engineers. They include twenty-two components and structures: PWR and BWR primary pressure boundary components, primary containments, feedwater lines, emergency diesel generators, and cables and connectors. The in-depth laboratory study of aging of these components is not feasible because of the large amount of funding and time required; however, a significant number of laboratory and in-field studies have been performed for these components and structures. The major element of the strategy adopted by INEL is to integrate, evaluate, and update the aging-related information from the completed and ongoing NRC, EPRI, and industry research programs and from several aging related databases. The other elements of the strategy include direct contribution from twenty-five experts having state-of-the-art aging-related knowledge of a particular component or structure and include peer-review of technical reports by the experts at the USNRC, national laboratories, and industry. The experts and INEL staff have worked together with experience in assessing age-related degradation of nuclear SSCs, have worked with the INEL staff to identify degradation sites and mechanisms, stressors, relevant field failures, and potential failure modes and to evaluate the current inspection, surveillance, and monitoring methods for each major component and structure. These results are presented in NUREG/CR-4731, Volumes 1 and 2. Figures 6.5 - 6.15 are concise summaries of the systematic and in-depth aging assessments that have been conducted. A similar strategy is followed to develop life assessment procedures for major components and structures and to evaluate emerging inspection, monitoring, and material testing techniques to characterize aging damage.



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**FIGURE D.2.** Application of NPAR Strategy to Understand and Manage Aging of Snubbers

**Case 4 - In-Plant Evaluation of Inspection, Surveillance and Monitoring Methods**

In-plant evaluation of inspection, surveillance, and monitoring methods (IS&MM) practiced at a representative pressurized water-reactor power plant was performed. The evaluation covered standard and advanced techniques in use and assessed the applicability of each technique for detecting and monitoring age-related degradation in reactor protection and 1E power system components. A comparison was made between current and advanced IS&MM to determine if the use of advanced methods is practical in the field and, if so, whether they provide information for trending aging that is not provided by current methods. The advanced IS&MM techniques included a state-of-the-art infrared thermography data acquisition and analysis system, the Redundant Instrument Monitoring System (RIMS™), and the Electronic Characterization and Diagnostic



(ECAD™) System. The evaluations determined that the advanced IS&MM investigated do provide information and methodologies for trending that is not currently provided by routine surveillance procedures.

### D.3 APPROACH III - IN SITU EVALUATIONS

Pacific Northwest Laboratory has performed several in situ evaluations of operating power reactor service water systems. These evaluations were done using a modeling approach to determine root causes. This approach was developed during the RES program to assess the effects of operating and maintenance methods and procedures and to assess the measurement techniques used to determine the thermal hydraulic capabilities of an active service water system. The need for these evaluations developed from 1) functional failure events in critical service water components, and 2) utility efforts to comply with generic letter 89-13, Service Water System Problems Affecting Safety-Related Equipment, July 18, 1989, which requires thermal performance verification of safety-related service water system components. Reports were prepared and delivered to Nuclear Regulatory Research, which presented an evaluation of the safety significance of operational aging on service water system thermal hydraulics. These in situ case studies affirmed both the methodology and the practical application of the NPAR approach to research on component degradation.

In situ assessments also were conducted on electrical circuits and selected components (e.g., cast stainless steel) at the Shippingport reactor (see Appendix E, Allen & Johnson 1990).

APPENDIX E

SHIPPINGPORT STATION AGING EVALUATION  
NRC CONTRACTOR PUBLICATIONS

## APPENDIX E

### SHIPPINGPORT STATION AGING EVALUATION NRC CONTRACTOR PUBLICATIONS

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11. ABSTRACT (200 words or less)

A comprehensive Nuclear Plant Aging Research (NPAR) Program was implemented by the U.S. NRC Office of Nuclear Regulatory Research in 1985 to identify and resolve technical safety issues related to the aging of systems, structures, and components in operating nuclear power plants. This is Revision 2 to the NPAR Program Plan. This plan defines the goals of the program, the current status of research, and summarizes utilization of the research results in the regulatory process. The plan also describes major milestones and schedules for coordinating research within the agency and with organizations and institutions outside the agency, both domestic and foreign.

Currently, the NPAR Program comprises seven major areas: 1) hardware-oriented engineering research involving components and structures; 2) system-oriented aging interaction studies; 3) development of technical bases for license renewal rulemaking; 4) determining risk significance of aging phenomena; 5) development of technical bases for resolving generic safety issues; 6) recommendations for field inspection and maintenance addressing aging concerns; 7) and residual lifetime evaluations of major LWR components and structures. The NPAR technical database comprises approximately 100 NUREG/CR reports by June 1991, plus numerous published papers and proceedings that offer regulators and industry important insights to aging characteristics and aging management of safety-related equipment. Regulatory applications include revisions to and development of regulatory guides and technical specifications; support to resolve generic safety issues; development of codes and standards; evaluation of diagnostic techniques (e.g., for cables and valves); and technical support for development of the license renewal rule.

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