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Residual Life Assessment of Major Light Water Reactor Components—Overview Volume 1

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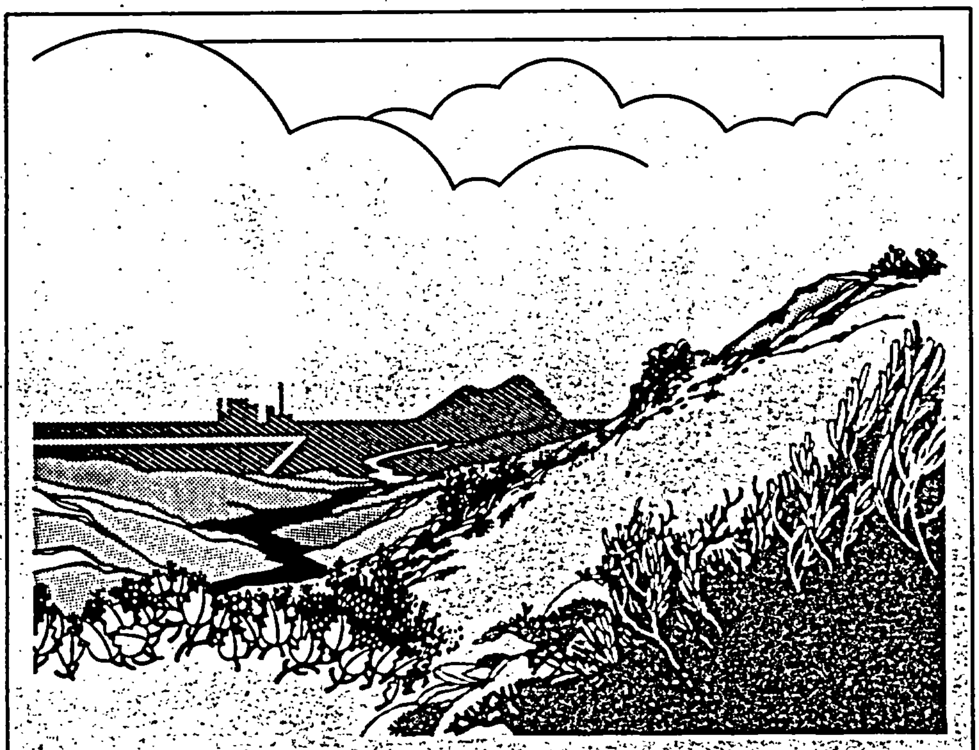

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RESIDUAL LIFE ASSESSMENT OF MAJOR LIGHT WATER REACTOR COMPONENTS—OVERVIEW VOLUME 1

V. N. Shah and P. E. MacDonald, Editors

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environment and cyclic loading. Likely degradation mechanisms include hydrogen embrittlement of the anchorages; corrosion of the liner, tendons, reinforcing steel, and metal shells, including possible microbiologically induced corrosion; and chemical reactions in the concrete. The potential failure modes are loss of prestress in the tendons, leakage of radioactive material caused by liner-concrete interactions, and loss of structural integrity, mainly because of corrosion. The ISI includes a tendon surveillance program, integrated leak rate tests, and visual inspection of surfaces. Major unresolved technical issues are a lack of aging related data for reinforced concrete and tendons, the need for improved inspection programs to identify and quantify degradation, and a better understanding of the potential impact of concrete-liner interactions in aged containments. A comprehensive and standardized ISI program is needed to identify and quantify degradation in reinforced concrete. Integration and evaluation of the available research results and information from older facilities on the degradation of reinforced concrete subjected to long-term exposure to elevated temperatures, radiation, and cyclic loading will support the development of an appropriate ISI program.

PWR Reactor Coolant Piping

The key degradation sites for PWR reactor coolant piping are the main coolant pipe nozzles, dissimilar metal welds, and cast stainless steel components. The reactor coolant piping is subjected to thermal and pressure loading caused by system operating transients. The major degradation mechanisms are low-cycle fatigue and thermal aging. The potential mode of failure is through wall leakage. ISI methods include surface and volumetric inspection. Two unresolved technical issues are a lack of accurate accounting of operating transients, and the development of a better understanding and assessment of the embrittlement of cast stainless steels because of thermal aging.

PWR Steam Generators

The key degradation sites in the recirculating steam generators are the inside tube surfaces at the U-bends, tube sheet, and tube supports; the outside surfaces at the tube-to-tube sheet crevice region; and the girth welds in the upper shell region. The corresponding sites for once-through steam generators are the outside surface of the tubes in the upper tube sheet region, and the welds attaching the thermal sleeves to the auxiliary feedwater inlets. The major degradation mechanisms include intergranular stress corrosion cracking (IGSCC), intergranular attack (IGA), pitting, wastage, denting, fretting, and thermal fatigue. The likely modes of failure are cracking, localized or uniform tube thinning, and wear out of tube material, which will ultimately lead to leakage of primary coolant to the secondary system and possibly to the outside environment. ISI methods include eddy-current testing and leak detection methods. Some of the steps taken to mitigate the effects of the various degradation mechanisms are using AVT chemistry on the secondary side, shot peening the U-bend and roll-transition regions of the tubes, using thermally treated Inconel 600 tube material and a quatrefoil design for tube support plate, and using 12% chromium ferritic stainless steel as tube support material. Additional work is needed to understand and model the corrosion mechanisms, and to better monitor the status of steam generator tube degradation.

Reactor Pressure Vessel Supports

The potential degradation sites for the neutron shield tanks and column supports are at the core horizontal midplane elevations, and for the cantilever supports they are in the active height of the core. These RPV supports are subjected to neutron irradiation, tensile stresses, operating temperatures, and a corrosive environment because of the presence of water. The major degradation mechanisms include neutron embrittlement, corrosion, and radiation damage to the lubricant used in the sliding foot assembly. The potential failure mode of the neutron tank, cantilever, or column supports is brittle fracture. Two remaining types of supports, i.e, the skirt and bracket types, are not likely to undergo catastrophic brittle failure because they are exposed to very little irradiation. However, the skirt support is subjected to fatigue because of the thermal- and pressure-induced expansions and contractions of the RPV during startup and shutdown. Currently, there are no requirements for ISI of RPV supports. Additional work is needed to develop a fracture toughness data base for RPV support steels irradiated at <232°C (450°F), determine the range of radiation environment conditions for support structures, and investigate the effects of actual radiation levels on lubricants used in the sliding foot assemblies and between RPV nozzles and supports.

BWR Reactor Pressure Vessels

The key degradation sites for BWR RPVs are nozzles, safe end welds, closure studs, and beltline region. BWR vessels are subjected to mechanical and thermal loads and neutron irradiation. The major degradation mechanisms are low and high cycle fatigue and neutron embrittlement. The potential mode of failure is ductile overload leading to a leakage. A surveillance program similar to the PWR programs is required to assess irradiation damage. ISI methods include volumetric inspection of weldments, studs, and threads. An unresolved technical issue is the need for close monitoring of nozzle fatigue usage.

BWR Recirculation Piping

The key degradation sites for BWR recirculation piping are the dissimilar metal welds at the safe ends, the cast austenitic stainless steel components, and the crevices at the shaft sleeves. The BWR recirculation piping is subjected to cyclic tensile stresses, an oxygen environment, high temperatures, and has sensitized heat-affected regions. The major degradation mechanisms are IGSCC, thermal fatigue, thermal embrittlement, and crevice corrosion. The potential mode of failure is a leakage through a crack in the piping. ISI methods include ultrasonic examination and use of moisture-sensitive tape. Two unresolved technical issues are the lack of an accurate accounting of operating transients, and the development of improved assessments of embrittlement because of thermal aging. Additional work is also needed to develop a better understanding of the effects of the hydrogen added to the recirculation piping loop to reduce the oxygen level in the coolant.

Current In-service Inspection Methods

Many of the standard NDE methods employed to satisfy ISI requirements were developed for the detection and qualitative assessment of fabrication-related defects. These methods are not entirely adequate for residual life assessment. Inspections for life assessment generally require greater detection reliability and a more quantitative determination of defects and accumulated damage than traditional ISI. The ISI methods generally practiced by the nuclear industry are visual examination, penetrant and magnetic particle testing, x-ray radiography, eddy-current testing, ultrasonic testing, and occasionally acoustic emission monitoring.

Visual examination is the most widely used NDE method. In most cases, it provides an indication that damage may have occurred but cannot directly quantify the amount of material damage. Penetrant and magnetic particle testing are used to improve the visibility of surface-connected flaws, and are therefore an extension of the visual examination method. X-ray radiography measures density variations that may be due to cracks, inclusions, porosity, voids, lack of bonding, and dimensional changes. The application of x-ray radiography is restricted because of the slow rate of examination. The use of single- and multifrequency eddy-current testing is generally limited to inspection of near-surface cracks in simple geometries such as PWR steam generator tubes. However, the standard eddy-current methods are not adequate to detect and characterize circumferential flaws caused by intergranular stress corrosion attack. Ultrasonic testing has been accepted as the most useful volumetric examination method, especially for inspection of welds and adjacent base material. The limitations of standard ultrasonic test methods are due to deficiencies in the available technology and human factors, such as operator boredom. Acoustic emission techniques are used to detect growing flaws in pressure vessels. The main advantage of the acoustic emission method is its use as a precursor to impending failure (rather than a method for flaw sizing required for residual assessment). Another useful application of acoustic emission monitoring is leak detection in pressure vessels. Among all the standard NDE methods, the eddy-current and ultrasonic methods are the most promising for making quantitative damage related measurements needed for residual life assessment.

The ISI of major nuclear power plant components is controlled by USNRC regulations and Section XI of the ASME code. Improvements in the ASME code NDE methodology are being made to detect flaws in piping, RPVs, containments, and steam generators. The current ASME code methodology is not adequate to assess the residual life of the major LWR components. More efforts are needed to develop field-usable NDE techniques and equipments. The major unresolved issues associated with current ISI methods are (a) the need for quantitative sizing of flaws for use in fracture-mechanics analyses, (b) the need for methods to inspect cast stainless steel components, and (c) the need for techniques to measure the degradation in mechanical properties during long-term service exposure.

Current Life Assessment Methods

Life assessment techniques include testing of surveillance specimens, monitoring of operational parameters, evaluating samples removed from service-exposed components, and predicting damage accumulation processes. The data from the surveillance programs used to assess the irradiation embrittlement of the RPV material show a good correlation between the brittle-to-ductile transition temperature shifts at 30 ft-lb (41 J) and the tensile yield strength. However, a correlation with a more relevant material property such as fracture toughness of the RPV steel is not well established. Monitoring of the operational pressures and temperatures may be used to determine the fatigue usage factors at critical locations in the primary loop. Metallographic and fractographic examinations and fatigue testing of samples removed from service-exposed components have been used to assess the structural integrity of LWR piping systems. Analytical prediction methods are required to estimate accumulation of damage during anticipated future operations. A fracture-mechanics approach based on probabilistic life assessment methodology can be used to estimate the reliability of pressure vessels and piping. Two major unresolved issues are (a) lack of confidence in the adequacy of the current models for material damage accumulation processes, such as irradiation embrittlement and fatigue crack initiation and growth under a spectrum of loadings, and (b) the unavailability of archival mechanical property data for comparison with the properties of the same material after service exposure, so that the degree of degradation can be assessed.

Emerging Methods for Inspection and Life Assessment

Recent interest in LWR plant life extension is encouraging the development of new methods for inspection and life assessment. Inspection methods are needed to accurately determine the size, shape, location, orientation, and type of both surface and internal flaws, (including microstructural phase changes), so that fracture mechanics approaches may be used for life assessment. Current NDE methods are not capable of detecting time-dependent changes in microstructural features, such as changes in the ferrite phase in austenitic-ferrite stainless steels, and the precipitate phases in stainless steel; or the radiation embrittlement of RPV steels. Such changes can be detected by replication, extraction, indentation hardness testing, x-ray diffraction, and electrochemical testing at assessable locations. Another potential method is the use of eddy-current to measure residual stress.

The synthetic aperture focusing technique (SAFT) for ultrasonic testing (UT) has been developed to provide enhanced visual images of flaws detected during inspection. While the initial results of the SAFT-UT application in the nuclear industry are encouraging, additional work is needed to fully qualify this technique. Another emerging method, a dc potential drop method, may be used to detect surface cracks on the inside of a pipe. Computer-aided techniques for eddy-current testing are also being developed to detect flaws in austenitic stainless steel pipe.

The USNRC is sponsoring programs using acoustic emission technology for on-line monitoring of crack growth in pressure boundaries, and leak surveillance in LWR systems. On-line monitoring appears to be a promising technique to detect fatigue-crack growth in RPVs and stress-corrosion crack growth in stainless steel piping. Initial evaluation results of the acoustic emission leak-detection technique are encouraging, but the method requires additional field validation.

The emerging methods for life assessment employ miniature test samples that can be removed from a component—with negligible damage to the component, on-line procedures for the calculation of damage, and improved models of material degradation and damage accumulation. Miniature specimen testing (MST) can provide a direct measure of the degree of aging. Miniature specimens can be valuable for surveillance testing where only a limited amount of test material is available and where space available for material irradiation is restricted. The main constraint to the application of MST is that the specimen must be large enough to be representative of the material from which it has been removed. MST may be employed to measure stress-strain response, fracture toughness, and fatigue cracking. MST also may be used to measure the through-thickness properties of thin steel plate, and to characterize the material properties near welds.

On-line damage and remaining life calculations may be performed for some component locations by monitoring key operating parameters on-line. For example, this approach can be applied to calculate fatigue usage factors for RPV nozzles during startups, shutdowns, and major operating transients when the nozzle temperatures are monitored during operation.

Finally, effective engineering models are required for the successful implementation of life assessment strategies. These models must provide a balance between engineering sophistication and practical utility so that good assessments can be made economically. Using simplifying assumptions and the results of the emerging improved inspection and monitoring methods, miniature specimen testing and on-line monitoring, these models should be capable of predicting the major, important features of material degradation.

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CONTENTS

RESIDUAL LIFE ASSESSMENT OF MAJOR LIGHT WATER REACTOR COMPONENTS—OVERVIEW VOLUME 1

1. INTRODUCTION

The United States was one of the first nations to use nuclear power to commercially generate electricity and, therefore, has some of the oldest operating commercial reactors. As U.S. light water reactors have matured, problems associated with time- or cyclic-dependent degradation (aging) mechanisms such as stress corrosion, radiation embrittlement, fatigue, and other effects have occurred and have raised questions about the continued safety and viability of nuclear plants and, in particular, about the integrity of the primary coolant pressure boundary. These problems have included cracked piping at boiling water reactors (BWRs), steam generator degradation at pressurized water reactors (PWRs), defective valves and relays and inadequate means for detecting and characterizing flaws.

At the same time, with a continually increasing demand for electricity and limited new generating capacity under construction, the U.S. electric utilities are motivated to keep their existing plants operating beyond the original design life at as high a capacity factor as possible. The economics of plant life extension are clearly favorable. Studies cosponsored by the U.S. Department of Energy (DOE) and the Electric Power Research Institute (EPRI) show that replacing any single nuclear plant component can easily be justified, if the life of the plant can be extended for a number of years. Extending the life of a 1,000-MW plant by 20 years is expected to realize a net present worth of about $1 billion.

Therefore, the potential problems of managing aging in older plants and the resolution of technical safety issues in consideration of the development of appropriate life extension criteria have become a major focus for the research sponsored by the U.S. Nuclear Regulatory Commission (USNRC). This is reflected in the following Policy and Planning Guidance (PPG NUREG-0885) provided to the NRC staff in 1986:

> "Requests for an operating license renewal are to be anticipated and will require advanced planning and analysis. The Commission intends to continue development of the policies and criteria to define requirements for operating license extensions to help assure that industry's efforts in this area are focused on the primary regulatory concerns."

An important part of the USNRC research effort is the Nuclear Plant Aging Research (NPAR) Program that is being conducted at several national laboratories, including the Idaho National Engineering Laboratory (INEL). One of the NPAR program tasks at the INEL is to develop the appropriate technical criteria for the USNRC to assess the residual life of the major LWR components and structures. These assessments will help the USNRC resolve certain safety issues and develop policies and guidelines for making operating plant license renewal decisions. Most of the effort for this residual life assessment task is focused on integrating, evaluating, and updating the technical information relevant to aging and license renewal from current or completed NRC and industry research programs. A five-step approach is being pursued to accomplish the residual life assessment task: (a) identify and prioritize major components, (b) identify degradation sites, mechanisms, stressors, potential failure modes, and evaluate current in-service inspection (ISI) methods, (c) assess current and advanced inspection, surveillance, and monitoring methods and evaluate maintenance programs, (d) develop residual life assessment models, and (e) develop criteria for license renewal. This report addresses the progress made toward gaining a qualitative understanding of the degradation mechanisms active in several light water reactor components and it represents a completion of the first step and partial completion of the second step of the approach described above.

Virtually all the major components and structures within a nuclear plant complex must be evaluated in a life-extension program. However, from the USNRC's perspective of ensuring the health and safety of the public, the first step in the residual life assessment task is to identify those major components that are critical to nuclear power plant safety during normal operation, off-normal conditions, design-basis accidents such as a hypothetical large-break, loss-of-coolant accident or the design-basis earthquake, or a severe accident. This report presents appropriate criteria for selecting the components and structures of interest and then ranks these components in order of importance. The next step is to integrate the available technical information so as to identify the degradation sites, stressors, degradation mechanisms, failure modes, and ISI methods associated with each of these major components. This report identifies the degradation sites and corresponding processes for seven major components mentioned later in this chapter. The remaining components will be addressed in a later edition of this work. Future work will be concerned with developing recommendations for quantitative residual life models.

A major factor affecting the life of the component is the amount of aging, i.e., degradation of performance over time, in the materials from which the component is fabricated. This is critical for those components that have a relatively small original safety margin. The degree of aging is often referred to as the damage state of the material. Material that has not been degraded by service would have zero damage, whereas a failed material would have 100% damage. Knowledge of the current damage state of the material is essential to assess the residual life because it establishes the initial conditions for the remaining life. ISIs are performed to measure the current state of damage. This report discusses the current nondestructive examination (NDE) methods used to detect and characterize defects in materials. This report also discusses certain new or emerging examination methods that are being successfully used in other industries and that could be used to quantify damage states and measure material properties in nuclear power plant components. With an adequate characterization of the damage state and a quantitative model, an accurate assessment of the residual life of selected major structural components will be possible.

In addressing this task on residual life assessment, major emphasis has been placed on integrating the available technical information on the relevant degradation mechanisms and applicable NDE methods. The sources for the technical information are technical reports from the USNRC and EPRI, technical workshops and conferences, American Society of Mechanical Engineers (ASME) Section XI meetings, technical seminars, technical journals, technical publications in Europe and Japan, technical discussions with experts, and contributions from subcontractor specialists with expert knowledge of specific major equipment or NDE activities. Several additional sources of information will be added to this list. One source that is being currently pursued is the plans and results of certain ongoing research activities at the national laboratories and other research organizations. Also, operating experience data, i.e., licensing event reports and NPRDS, including information on operating transients and NDE activities taking place at nuclear power plants will be used.

This report is organized into three major parts. The first part, Chapters 2 through 7, discusses major PWR components. The second part, Chapters 8 through 10, discusses major BWR components. The third part, Chapters 11 through 14, discusses NDE methods.

Chapter 2 presents the criteria used to select the major PWR components for this residual life assessment task. These criteria are also used to select the major BWR components. Chapter 2 ranks the selected major components according to their relevance to plant safety. Chapters 3 through 6 discuss the aging of the four highest ranked PWR components: the reactor pressure vessel, the containment, the reactor coolant piping, and the steam generators, respectively. The discussion for each component includes a description of the component, the stressors acting on the component, the degradation sites and their rankings, degradation mechanisms active in the component, potential failure modes of the component during normal operation and transient and accident conditions, and current ISI methods used for the component. A prioritized list of degradation sites and associated degradation mechanisms, potential failure modes, and ISI methods are presented in a table at the end of each section. The discussion for the other components follows a similar sequence. Chapter 7 discusses the aging of the reactor pressure vessel supports in both PWR and BWR plants. The major BWR components are identified and prioritized in Chapter 8. Chapters 9 and 10 discuss the aging of the BWR reactor pressure vessels and recirculation piping, respectively. Chapter 11 discusses the standard ISI methods. Chapter 12 discusses the current NDE activities in the ASME Section XI committees. Chapter 13 discusses the current life assessment methods. Chapter 14 evaluates the applicability of advanced NDE and life assessment methods employed in the fossil power plant and petrochemical and aerospace industries as well as those being developed for LWR applications. Finally, Chapter 15 presents the conclusions and recommendations for further work to assess the residual life of selected LWR components.

2. RANKING OF MAJOR PRESSURIZED WATER REACTOR COMPONENTS

V. N. Shah

Residual life assessment of the major components is of primary interest for continuing safe operation and life extension of a nuclear power plant. This chapter identifies the major components of interest in commercial pressurized water reactors (PWRs). In identifying these components, it is assumed that the lifetime of the sensors, controls, batteries, certain types of pumps, valves, motors, etc. is much shorter than 40 years and these smaller, less expensive components will be maintained, refurbished, and/or replaced frequently. Therefore, the residual life of these small components will not significantly affect the life extension of the plant and is not addressed in this report.

The major components reported here are identified and prioritized according to their relevance to plant safety only. The industry-sponsored Surry-1 pilot plant project has also identified components for plant life extension studies.[1] However, the identification and prioritization criteria used by industry pilot project were based on plant safety, reliability, and cost.

2.1 Key PWR Components

The selection of the key components is based on two main safety criteria: to contain the release of fission products that may take place during an accident, and to maintain an acceptable low level of radioactivity in the containment during normal operation. There are several barriers to the release of fission products. Two of these barriers are associated with the fuel rods. Because these rods are replaced several times during the lifetime of a reactor, they are not included in the selected components. The remaining two barriers are the pressure boundary components and the containment. Therefore, the pressure boundary components, containment, and their supporting structures are included in the list of selected major components. The control rod drive mechanism (CRDM) is included in the list because its failure may lead to an anticipated transient without scram (ATWS) or reactivity insertion accident. The safety-related cables and connectors are included because they provide signals related to plant condition to the sensors that control the operation of the safety systems. The failure of the cables and connectors because of aging may lead to common-cause failure of a number of systems. The emergency diesel generator is included in the list because it is required for plant safety during a station blackout event. The reactor pressure vessel (RPV) internals are included in this list because their failure may prevent control rod insertion or cause fuel failure. The RPV supports and biological shields maintain the level of radiation in the containment at an acceptable level and provide support to the RPV and primary coolant piping, and therefore they are included in the list.

Major PWR Components

1. Reactor pressure vessel
2. Containment and basemat
3. Reactor coolant piping and safe ends
4. Steam generators
5. Reactor coolant pump body
6. Pressurizer
7. Control rod drive mechanisms
8. Cables and connectors
9. Emergency diesel generators
10. RPV internals
11. RPV supports and biological shields.

The RPV is ranked Number 1 because it is the most critical major component as far as plant safety is concerned. It is one of the major components of a nuclear power plant for which there is no redundant member. The catastrophic failure of a pressure vessel could lead to a rapid core meltdown, generating high pressure and temperature loadings for which light water reactor containments are not designed. Therefore, the RPV is ranked higher than the containment. The vessel is subjected to high system pressures, temperatures, neutron irradiation, and cyclic fatigue. The most likely degradation sites are the circumferential and axial weldments in the beltline region of the vessel. Other possible degradation sites of concern are the hot-leg and cold-leg nozzles, the weldment at the lower shell to bottom head junction, the vessel flanges and studs, and the instrument penetrations in the bottom head. The most important potential degradation mechanism is neutron embrittlement. The factors affecting neutron embrittlement of RPV steel are fast neutron fluence, irradiation temperature, and the chemical composition of the steel and weld metal (mainly Cu and Ni content).[2,3] The fast neutron fluences and the chemical composition of the weld metal vary circumferentially and through the thickness of the RPV wall. The irradiation effects on the RPV steels are well documented, required to be considered in operation and test limits, and monitored by surveillance programs during service. A less critical degradation mechanism is fatigue. However, the damage caused by fatigue is irreversible, while irradiation damage of pressure vessel steels may be reversible.

The containment and basemat are ranked higher than the remaining components because they are the major barriers protecting the public during an accident from released fission products. Under a severe accident scenario, it is possible that the containment internal pressure can rise to a value that may cause leakage through the containment wall. LWR containment structures may be divided into three types: prestressed concrete containments, reinforced concrete containments, and steel cylinder/steel sphere containments. The most likely degradation site for a prestressed concrete containment is the posttensioning system, consisting of tendons and anchors. The potential degradation mechanisms for the posttensioning system are time-dependent relaxation of tendon material, and hydrogen embrittlement and stress corrosion cracking of the heat treated anchor heads.[4] Corrosion of the tendon material is also possible. The liner constitutes the weakest element in the prestressed and reinforced containment structures during a design basis accident (DBA) and severe accidents.[5] A liner-concrete interaction may take place at locations of major concrete cracks and distortions and introduce highly localized strains in the liner. This strain concentration may cause localized rupture of the liner that will lead to leakage from the containment at a relatively early time in a pressurization transient. This leakage to the environment represents the most likely failure mode for the dry and subatmospheric types of containment but not for the dual containment, and it is more likely in the reinforced structures than in the prestressed structures. Prestressed structures see relatively minor cracking until nearly twice the design pressure, but reinforced structures develop widespread cracking at about 75% of the design pressure. The relevant degradation modes affecting the concrete material properties are deterioration because of aggressive environments and internal chemical reactions. The aggressive environments include nuclear heating, extensive cycling of wetting/drying and freezing/thawing of the exterior walls, acid rains and sulfate-bearing ground water, and carbonation.[6,7] Carbonation is the process by which the alkaline calcium hydroxide present in the concrete is neutralized by carbon dioxide and atmospheric moisture.[8] Cracked concrete can provide a direct path of water through or near to the rebars that may lead to corrosion of reinforcing steel and subsequently to additional cracking and spalling of the concrete. The internal chemical reactions include alkali-aggregate, cement-aggregate, and carbonate-aggregate reactions. The most likely degradation sites for steel cylinder/steel sphere containments are the base metal and the weldments in the containment wall. The major degradation mechanism is the corrosion of the steel walls.

The four components representing the pressure retaining boundary are ranked right below the containment and vessel because they constitute the first barrier to the release of the fission products following the failure of fuel rod cladding during an accident. These components are the reactor coolant piping and safe ends, the steam generator, the reactor coolant pump body, and the pressurizer. The failure of any of these four components may lead to a large-break or small-break loss-of-coolant accident (LOCA) that may challenge the integrity of the containment by imposing high thermal and pressure loadings on it. The reactor coolant piping and safe ends are ranked third, i.e. the highest among these four components, because the impact of a failure is severe. A break of the reactor coolant piping will impose significant mechanical loads on the component supports and other structural members and may damage them. The weldments at the safe ends have dissimilar metal welds, and are the most likely locations for degradation. The branch nozzles and the cast stainless steel components, i.e. elbows and T-connections, etc., in the reactor coolant piping are also possible locations for degradation. The potential degradation mechanisms are thermal fatigue and thermal embrittlement.

The steam generator is ranked next to the reactor coolant piping because the rupture of its tubes will provide a passage for primary coolant to outside the containment by way of the relief valves. In addition, a major steam generator tube rupture event is of serious nature because of the rapid depressurization of the primary coolant system and the complications that result in terminating the pressure transients that ensue. In fact, relatively small tube rupture accidents have taken place at several nuclear power plants,[9] and the steam generators in the operating PWRs have been plagued by failures because of mechanical and chemical problems. The majority of problems have been associated with the recirculating type units manufactured by Westinghouse and Combustion Engineering. However, the once-through units manufactured by Babcock & Wilcox have also experienced some operational difficulties.[10] The locations of the degradation sites within a recirculating type steam generator include: the outside surfaces of the tubes near the support plate in the region of the cold leg,[1] the inside surface of the Inconel tubing in the region of the hot leg,[11] the divider plate, the tube support plates, the feedwater nozzle, the J-tubes attached to the feedring,[12] and the upper shell to transition cone girth welds. The J-tubes are normally fabricated from carbon steel and have been observed to degrade through wall thinning and perforation by a corrosion-erosion mechanism. The J-tubes in the recirculating type units manufactured by Westinghouse have been replaced, and therefore they probably will not have a significant impact on the life extension of those steam generators. A potential degradation mechanism for the girth welds is corrosion fatigue.[13] The normal operating water level for the steam generators is in the vicinity of those welds and the normal oscillations of the water level may impose cyclic thermal loads on the welds. Cracks have been discovered in the girth welds of the Surry-2 and Indian Point-3 steam generators.

The locations of the degradation sites within a once-through steam generator (OTSG) include the secondary face of the upper tube sheet and the uppermost tube support plate in tube rows adjacent to the inspection lane.[10] Another likely degradation site is the weld at the auxiliary feedwater nozzle thermal sleeve in the OTSG. The cracking of this weld will increase the potential for thermal shock to the OTSG shell. The potential degradation mechanisms for steam generator tubes are wastage, denting, intergranular stress corrosion cracking, intergranular attack, erosion corrosion, pitting, and fretting. The use of all-volatile treatment of the secondary coolant has significantly reduced the damage due to wastage and denting in the recirculating type steam generators.[14] The potential degradation mechanism for the divider plate and feedwater nozzle is thermal fatigue.

The massive size of the reactor coolant pump casing provides primary protection to the pressure retaining components within the reactor coolant system and containment building from missiles generated by internal failure of pump elements (e.g. impeller parts, shafts, fasteners). Therefore, it is ranked higher than the pressurizer. The reactor coolant pump casings in the PWR systems are thick (8 in. or greater) and constructed of cast austenitic steel. Because of the casing size, the pumps manufactured by Westinghouse are welded by the electroslag process that introduces high residual stresses in the heat-affected zones near the weldments.[15] The pump is powered by a large electric motor (4000 hp and larger) to maintain the coolant flow through the reactor vessel, and the pump casing is subjected to the high vibratory forces of the rotating pump impeller assembly. The vibratory forces, in combination with operating stresses, fluid dynamic cyclic loadings, and the primary coolant environment, may contribute to flaw growth. The potential degradation mechanisms for the pump casing are corrosion fatigue and thermal embrittlement.[16] The other possible locations for degradation sites are flange and seal housing boltings. Corrosion fatigue is the potential degradation mechanism for the boltings.

The pressurizer is the last key component that represents the primary coolant pressure retaining boundary. A surge line connects the pressurizer to the hot leg, while a spray line connects it to the cold leg. The surge and spray nozzle connections at the pressurizer vessel are subjected to cyclic temperature changes resulting from the transient conditions in the reactor. The spray nozzles are subjected to larger temperature changes than the surge nozzle. Thermal fatigue is the potential degradation mechanism for these two nozzles. The spray head, upper shell barrel, and seismic lugs are the other possible locations for degradation. The potential degradation mechanism for the spray head is erosion, and for the shell barrel and seismic lugs it is fatigue. Another potential degradation mechanism for the spray head is thermal embrittlement, if it is made of cast stainless steel.

The next three components play critical roles in mitigating LWR operational transients and accidents. The control rod drive mechanism is ranked highest among these three components because its failure may lead to an ATWS, or a reactivity initiated accident (RIA), or a small-break LOCA. The key subcomponents that may degrade are the drive rod assembly and the control rod pressure vessel. Electrical failure caused by closed contacts in a switch may result in a control rod withdrawal event leading to a RIA. Mechanical binding of the latch mechanism may result in the failure of the scram system and lead to an ATWS accident. The pressure vessel houses the latch assembly and drive rod assembly, and provides a pressure barrier for the CRDM between the primary coolant and containment. Failure of the control rod drive pressure vessel may result in a rod ejection accident along with a small-break LOCA. Eleven failures of pressure vessels and two failures of drive rod assemblies are reported in the data base that covers the information from 31 Westinghouse-designed, operating plants up to and including 1983.[17] The potential degradation mechanism for those pressure vessels having cast stainless steel latch housings, is thermal embrittlement.[1] Further research is required to determine the impact of the thermal embrittlement on the residual life of the CRDM. Wear is the most likely degradation mechanism for the drive rod assembly.

Safety-related cables and connectors are active during normal operation of the reactor, while the emergency diesel generators are needed if offsite power is lost. Therefore, the safety-related cables and connectors are ranked higher. The key degradation sites are the cable insulation, especially in the higher temperature regions of the containments, and the connectors. Cable insulation becomes brittle because of thermal aging, radiation, and oxidation, and may break during a seismic event causing electrical shorts. Cables may also fail by creep shortout, which occurs when a mechanically-stressed cable is bent over a corner and the conductors are pulled toward each other or toward a grounded electrode through insulation softened by aging.[18] Certain types of connectors have inserts that are made of carbon, which can become embrittled and fracture because of aging. The Westinghouse-designed reactor, San Onofre-1, which has been in operation since January 1968, has experienced the problem of embrittled connectors.[11] Moisture intrusion has caused corrosion of some connector material.

The emergency diesel generators must provide backup power to operate critical reactor safety equipment in the event of a loss-of-offsite power to the emergency bus bars. The failure of the diesel generators to provide backup power will lead to a station blackout transient. A station blackout transient followed by the failure of the steam-driven emergency feedwater pump to provide cooling water to the steam generators and the absence of operator intervention, is identified as a dominant core melt sequence by the recent USNRC reactor risk study.[19] The diesel generators have instrumentation and a control system that should start them automatically on loss-of-offsite power, low reactor water level, high contamination level, or other emergency condition signals, and automatically apply the required load. A review of the various failure modes of the diesel generators indicates that the governor in the instrumentation and control system is the subcomponent most susceptible to aging degradation.[20] The other subcomponents susceptible to aging degradation are the cooling system pumps and piping, the fuel system injector pumps and engine piping, and the turbocharger. The potential degradation mechanisms are metal fatigue and loosening of the fasteners.

The main function of the reactor internals is to provide orientation and support for the reactor core, and guide and protect the reactor control rod assemblies. It also provides a passageway, support, and protection for any in-vessel instrumentation. One of the potential failures of the RPV internals, i.e. baffle jetting, has led to fuel rod cladding degradation and disbursement of fuel into the coolant in certain reactors. Failure of the RPV internals also may relocate fuel away from the control rods or prevent the control rods from inserting properly and lead to an operational transient without scram. The key RPV internal components susceptible to aging degradation are the lower core plate, the baffle-former assembly, the upper support column bolts, the control rod guide tube sheaths and support pins, the thermal shield bolts, the core barrel bolts, the in-core instrument nozzles, and the flux thimble tubes.[1] Thermal shield and core barrel bolt failures have occurred in a number of older Westinghouse plants. Twenty-one out of 54 in-core instrument nozzles in the Oconee-1 RPV had been cracked and broken off in the region of the weld. The loose broken pieces of the in-core instrument nozzles subsequently caused extensive damage to the tube ends and to the tube-sheet welds in the upper head of one of the two steam generators.[21] It should be noted that some of the past RPV internal problems were caused or aggravated by original design errors. The potential degradation mechanisms are neutron embrittlement, stress-corrosion cracking, mechanical wear, low- and high-cycle fatigue, and stress relaxation. The low-cycle fatigue is caused by the loads due to changes in power levels, vessel inlet and outlet temperature differences, and coolant pressures and flow rates. The high-cycle fatigue is a result of the flow-induced vibrations. The control rod guide tube sheath and support pins, and flux thimble tubes are locations that may experience significant mechanical wear.

The last major components are the RPV supports and biological shields. There are several different designs of RPV supports and biological shields employed in the commercial nuclear power plants. One designed by Stone & Webster includes a sliding foot assembly and a neutron tank. The sliding foot assembly supports both the vertical and tangential loads of the RPV and its permanently lubricated foot slides radially to allow free thermal expansion of the RPV. The neutron tank supports the reactor and, in addition, stores water in the annular space between the RPV and the tank wall and provides shielding against neutrons. The sliding foot assembly is made of a maraging steel and coated with Heresite, an air-drying phenolic coating. The potential degradation mechanism for the sliding foot assembly is the stress-corrosion cracking of its threaded parts.[1,9] The potential degradation mechanisms for the neutron tank are neutron embrittlement and corrosion at the inside surface of the tank walls.

2.2 Summary, Conclusions, and Recommendations

The major PWR components are selected and ranked such that the release of the fission products that may take place during an accident is contained. The RPV has been identified as the most important component in PWR plants. This means that the failure of a reactor pressure vessel would have a very significant impact on the safety of the power plant. The other key components, according to their ranking, are containment, reactor coolant piping, steam generators, reactor coolant pump body, pressurizer, control rod drive mechanisms, cables and connectors, emergency diesel generator, RPV internals, and RPV supports. The insights and results presented in this chapter are based on past experience as reported in the available literature. These results will be updated as more data are made available.

Table 2.1 summarizes the aging related information on the key PWR components discussed in this section. It lists the components and gives reasons for their ranking. Table 2.1 identifies the most likely degradation site and other degradation sites for each component. Table 2.1 also lists the most likely degradation mechanism and other potential degradation mechanisms.

The major components reported in this section were selected and prioritized according to their relevance to plant safety only. These criteria are different than the ones used in the industry pilot study. Therefore, the ranking of the major components presented in this section is somewhat different than the one in the pilot study. However, all major components identified in this section were also ranked high in the industry pilot study. Detailed discussions of the possible degradation processes of PWR reactor pressure vessels, PWR containments and basemats, reactor coolant piping, steam generators, and RPV supports are presented in Chapters 3 to 7, respectively.

Table 2.1. Key PWR components for residual life assessment

Rank	Component	Reasons for Ranking	Degradation Sites (most likely, others)	Degradation Mechanisms (most likely, others)
1.	Reactor pressure vessel	Severe safety impact of catastrophic failure of an embrittled vessel	Weldments in the beltline region, hot-leg and cold-leg nozzles, weldment at the lower shell to bottom head junction, vessel flanges and studs, instrument penetrations in lower head	Neutron embrittlement, corrosion fatigue, thermal fatigue
2.	Containment and basemat	Public protection during an accident	Posttensioning system (tendons and anchors), concrete, metal liner, reinforcing steel	Hydrogen embrittlement and SCC of anchor heads, corrosion and relaxation of tendon material, environmental degradation of concrete, corrosion of reinforcing steel and liner
3.	Reactor coolant piping and safe ends	Severe safety impact of a large break	Weldments at the safe ends, branch nozzles, cast stainless steel components (elbows and t-connections), complete primary loop piping in new Westinghouse plants	Fatigue, thermal embrittlement
4.	Steam generator	Tube rupture will provide a passage from the primary system directly to the environment for the primary coolant, relatively poor operating experience	Inside tube surfaces at U-bends and tube sheet, outside surfaces at tube-to-tube sheet crevices, divider plate, tube support plate, feedwater nozzle, girth weld	Wastage, denting, IGSCC, pitting, fretting, IGA, thermal fatigue, corrosion fatigue
5.	Reactor coolant pump casing	Primary protection from any internal failure of pump elements (impeller parts, shafts), primary coolant pressure boundary	Casing wall, flanges, seal housing bolts	Corrosion fatigue, thermal embrittlement
6.	Pressurizer	Primary coolant pressure retaining boundary	Surge and spray nozzles, spray head, upper shell barrel, seismic lugs	Thermal fatigue, erosion, thermal embrittlement
7.	Control rod drive mechanism	Failure may lead to a reactivity initiated accident, an operational transient without scram, or a loss-of-coolant accident	Drive rod assembly, control rod pressure vessel	Wear and thermal embrittlement

Table 2.1. *(continued)*

Rank	Component	Reasons for Ranking	Degradation Sites (most likely, others)	Degradation Mechanisms (most likely, others)
8.	Safety-related cables and connectors	Active during normal operation in mitigating operational transients and accidents	Cable insulation, inserts in connectors	Thermal aging and creep of insulation, thermal embrittlement and corrosion of connectors
9.	Emergency diesel generator	Needed to operate critical safety equipment in the event of a loss of offsite power	Governor in the instrumentation and control system, cooling system pumps and piping, fuel injector pumps, turbocharger	Fatigue and vibrations
10.	Reactor internals	Failure may lead to dispersement of fuel into the coolant, or a reactivity-initiated accident	Lower core plate, baffle-former assembly, upper support column bolts, control rod bolts, control rod guide tube sheaths and support pins, thermal shield bolts, in-core instrument nozzles, flux thimble tubes bolts, in-core instrument nozzles, flux thimble tubes	Neutron embrittlement, wear, high cycle fatigue, SCC, relaxation
11.	RPV supports	Failure will challenge the integrity of the primary coolant piping and the small lines connected to the RPV. The neutron shield tanks maintain acceptable levels of radioactivity in the containment	Inside surface of the support structure (neutron tank or column support) at the core horizontal midplane level, lubricant in sliding foot assembly of neutron tank support	Neutron embrittlement, corrosion, degradation of lubricant

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3. PRESSURIZED WATER REACTOR PRESSURE VESSELS

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3.1 Description

The design of pressure vessels for commercial pressurized water reactors (PWRs) is very similar for the different domestic nuclear steam supply system vendors: Westinghouse (W), Combustion Engineering, Inc. (CE), and Babcock & Wilcox Co. (B&W). Both CE and B&W manufactured their own vessels, while W procured vessels from CE, B&W, Chicago Bridge and Iron (CB&I), or Rotterdam Dockyard Co. The vessels for Prairie Island, Units 1 and 2, were manufactured by Societe des Forges at Ateliers du Cresot. Also, at least one B&W vessel was completed by Rotterdam Dockyard Co. for W, and this vessel is part of the Surry-1 plant that is being studied in detail for life extension under Electric Power Research Institute (EPRI)/Department of Energy (DOE) funding. Vessels were designed and built in accordance with the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III[1] at the time of fabrication, except for the earliest vessels that were constructed before Section III existed. These earlier vessels were built to ASME Boiler and Pressure Vessel Code, Sections I or VIII. Different versions or updates to the Code apply to different vessels depending upon the vintage. The structural integrity of the Sections I and VIII vessels has been judged to be comparable to Section III vessels of approximately the same era.[2] Section III requires more limiting nondestructive examination of welds so that the probability of manufacturing defects is small.

The CE, early B&W, and CB&I production vessels used steel plates formed and welded to produce the vessel structure. The earliest commercial reactor was Yankee-Rowe (operational in 1961), which had a W vessel made by B&W of SA302B steel. SA302B was used in several early vessels until SA533B-1 steel plate became the industry standard. (Nickel-modified SA302B is essentially the same composition as SA533B-1). Automatic submerged arc welding was typically employed for both longitudinal and circumferential welds. The materials consumed in this welding process are a manganese-molybdenum-nickel filler wire and a granulated flux that minimizes atmospheric contamination and provides ingredients to form a slag to remove oxides during the welding process. The type of flux material is important because the mechanical properties of the weldment can differ depending upon what flux is used. B&W welds employed a Linde 80 flux that typically gives lower values of upper-shelf Charpy V-notch properties than other fluxes. CE and CB&I used Linde 0091, 1092, and 124 fluxes; those three fluxes produce similar mechanical properties. The Linde flux designations are proprietary codes used by the Linde Division of Union Carbide Corporation. Some shielded metal arc (SMA) welding may be used for complex geometry welds (nozzles), fit-up, backwelding, and repairs. The SMA electrode is a wire coated with a bonded flux (which provides a protective arc environment); the welding current is automatically controlled, but the arc voltage and welding speed are manually controlled. An E8018 electrode is typically used for SMA welds by all fabricators. Minimum preheat and interpass temperatures (∿300°F or 149°C) are controlled, and intermediate postweld heat treatment may be employed. All vessel welds are postweld heat treated to reduce residual stresses.

For later B&W vessels, an SA508-2 forging steel was used for fabrication. The main advantage of the forging is that the entire ring section is made of one piece, eliminating the need for longitudinal weld seams. The W vessels are only slightly modified versions of the CE and B&W vessel designs.

W has three slightly different versions of vessels, depending on whether a two-, three-, or four-loop plant was built; the number of loops dictate the overall size (diameter) of the vessel as well as the number of nozzles. B&W has three slightly different versions with the main difference being the change in fabrication from plate to forging material. The CE vessels change slightly in terms of diameter, but a more significant change is that the newer System 80 design is larger to reduce neutron exposure and has fewer welds.[3] Additionally, the nonsystem 80 CE-design vessels have instrumentation nozzles on the top head; all others (including the System 80) have bottom-head instrumentation.

A typical reactor vessel section is shown in Figure 3.1 to illustrate the important areas of concern. These regions will be discussed in more detail later. All vessels are lined with a stainless steel cladding (usually Type 308 or 309 stainless steel) that is applied in one or two layers by either multiple wire or strip cladding SMA processes.[4] The cladding

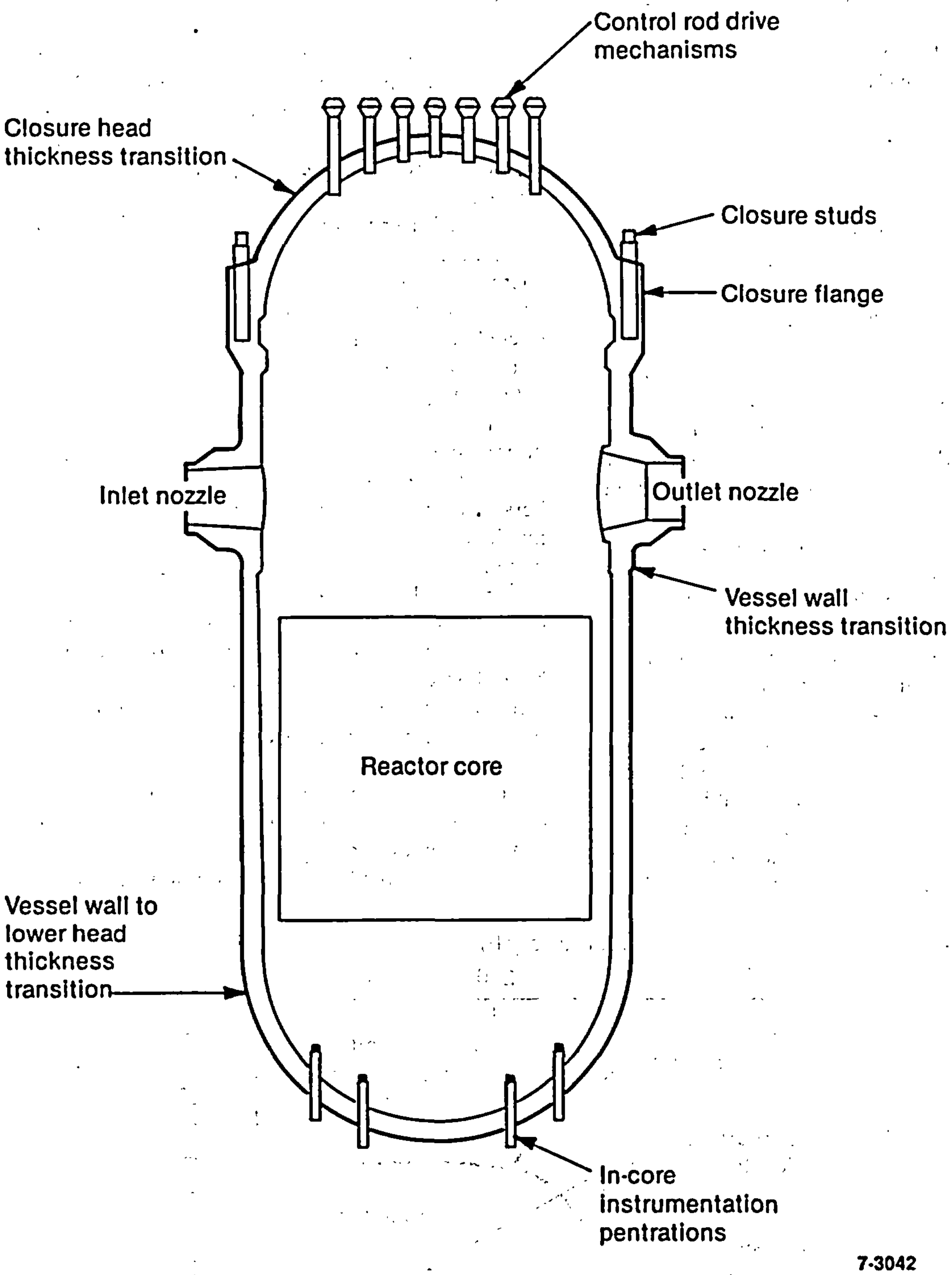

Figure 3.1. Typical PWR vessel showing important degradation sites.

thickness can vary, but is typically around 0.2 in. (5 mm).

Many of the issues described in the next subsections are also directly applicable to the discussions of Chapters 5 and 9 on primary system piping and BWR RPVs, respectively.

3.2 Stressors

The radiation environment is a major stressor because it causes a degradation in mechanical properties. The other primary stressors are mechanical pressure loads during operation, periodic thermal transients that induce through-wall bending moments, dead weight loads, residual stresses, and other abnormal loadings that can cause damage (such as pressurized thermal shock). Other stressors that can be important are water chemistry and mechanical contact that result in wear.

The effects of neutrons from the nuclear reaction process bombarding the inside wall of the reactor vessel are a serious issue for maintaining the integrity of the vessel. As a result of irradiation exposure, the ferritic steel strength increases with an attendant decrease in ductility. In terms of the Charpy V-notch (CVN) energy properties, the ductile-brittle transition temperature increases to higher temperatures as shown in Figure 3.2 (below the transition temperature, the material behaves in a brittle manner and its fracture requires only little energy, whereas above the transition temperature, the material behaves in a ductile manner), and the upper shelf level decreases in magnitude as the irradiation exposure increases (upper shelf level in the CVN energy versus temperature curve represents the fracture energy that corresponds to approximately 100% ductile shear and remains approximately a maximum for the upper range of test temperatures). The fracture energy of the pressure vessel steel thus changes continuously with radiation exposure at a given temperature. Therefore, 10 Code of Federal Regulations (CFR) 50 has established a criterion that defines the transition temperature as the temperature at the fracture energy level of 30 ft-lb (41 J). Monitoring of the condition of the reactor beltline region is conducted through the use of surveillance specimen testing, which is covered later in Section 3.4.1.

Operating transients that occur with regularity were used in the original ASME Section III fatigue design analysis. Typical normal, test, and upset transients are given in Table 3.1. The actual numbers of occurrences of these transients are usually much less since conservative numbers were estimated for the design. Other transients that are abnormal also are important since the fatigue design did not consider their occurrence. These events, such as a main-steam-line break or a small-break loss-of-coolant accident, can create a potential challenge to the vessel integrity (but not because of frequent occurrences).

Not only are many of the number of design transients unlikely to occur, but the magnitude of actual events are not generally as severe as those designed against. For example, plant operating procedures generally limit the heatup and cooldown rates to a

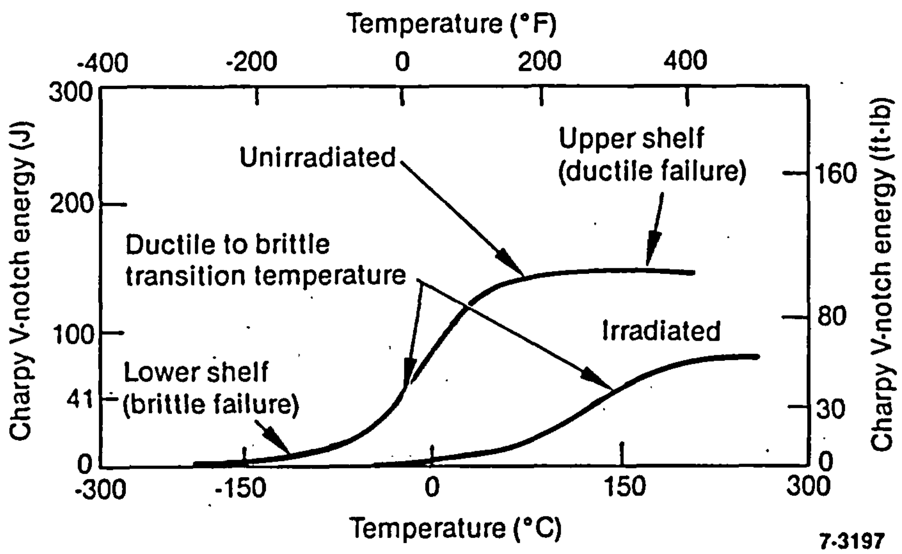

Figure 3.2. Effect of irradiation on the Charpy impact energy for a nuclear pressure vessel steel.

Table 3.1. Typical plant transients and assumed design occurrences[5,6]

Transient	Number of Events
Plant heatup at 100°F (38°C)/h	500
Plant cooldown at 100°F (38°C)/h	500
Plant loading at 5% full power/min	15,000
Plant unloading at 5% full power/min	15,000
Step load increase of 10% full power	2,000
Step load decrease of 10% full power	2,000
Reactor trip from full power	400
Loss of flow and abnormal loss of load	80
Loss-of-secondary pressure	5
Hydrotest to 3125 psig (21.55 MPa), 400°F (204°C)	10
Operating basis earthquake	200
Normal plant variation [100 psi and 10°F (-12°C)]	$>10^6$

maximum of 50°F/h (28°C/h), and actual practice reduces the rate even lower. Therefore, the original fatigue design is generally very conservative.

The primary transients leading to fatigue damage (in terms of the cumulative fatigue usage factor used in Section III of the code) are plant heatup/cooldown, plant loading/unloading, reactor trips, loss of flow, and abnormal loss of load. Other transients, such as inadvertent safety injections, can also contribute to fatigue damage even though they are not explicit design transients. If these transients can be controlled and properly monitored to indicate the actual level of loadings, estimates of the actual fatigue usage can be obtained. This issue is discussed further in Section 3.4.2.

The other issues of water chemistry and mechanical wear are less important. The water chemistry issue pertains mainly to the possibility of corrosion-assisted fatigue-crack propagation if a crack were to exist that penetrates the inner cladding layer and extended into the ferritic steel vessel material. Fatigue-crack growth rates for ferritic steels in water environments at operating temperatures may be an order of magnitude higher than those measured in air environments at the same temperatures.[7] The mechanical wear problem concerns surfaces that contact and periodically move relative to each other. The action of the forces between the surfaces creates potential wear of the surfaces that can lead to eventual inoperability.

3.3 Degradation Sites

The sites of highest degradation depend on the degradation mechanisms. Obviously, the sites of degradation relative to radiation embrittlement are in or near the reactor beltline region. Therefore, the welds within that region become possibly the weakest link because the welds are more likely to contain defects that can become cracks. Additionally, the problem exists of higher copper (and nickel) content in many of the older vessel welds leading to higher radiation damage sensitivity. The steel filler material used in the fabrication of older submerged arc welds was copper plated to enhance electrical contact during welding and resistance to corrosion during storage. The copper content in the welds results directly from the copper plating and the residual copper in the consumable steel filler material. It is unfortunate that the effects of copper (and nickel) were not recognized to be detrimental to neutron exposure resistance earlier. Appendix G to 10 CFR 50 defines the beltline region as including all shell material directly surrounding the effective height of the fuel element assemblies plus an additional volume of shell material both below and above the active core, with a predicted reference temperature shift (ΔRT_{NDT}) at the end of service life of 50°F (>28°C). In some earlier CE vessels, (especially considering license renewal) the 10 CFR 50-defined beltline region could even include the nozzles, the nozzle welds, and the nozzle shell. The base metals should not be ignored because the copper content in older plates and forgings was not controlled to a minimum level; however, there appears to be less radiation embrittlement in base materials as compared to welds with the same copper/nickel concentrations. Also it is possible that flaws can exist in the vessel cladding which could potentially lead to flaws in the ferritic vessel wall.

In the case of fatigue, most of the problems involve geometric discontinuities or changes. Informal review of CE-designed vessels[a] reveals the following order for the areas of relatively higher usage factors: closure head studs (inside bottom surface), outlet nozzles (two per vessel), inlet nozzles (four per vessel), and instrumentation nozzles (penetrations).

The instrumentation nozzles for the older CE design (at the top) did not have high fatigue usage factors, but the System 80 design (at the bottom) does. Other areas considered include the closure head itself and the vessel flange, control rod drive mechanism (CRDM) housings, vessel supports (at the nozzles), vessel wall transitions, vessel wall to bottom head juncture, surveillance holders, core stabilizing lugs, and a flow baffle. The CE design for integrally welded surveillance holders is unique because none of the other designs allow attachment to the vessel inner wall (cladding surface). The CE surveillance holders are made of a nickel-based alloy welded to the stainless steel cladding. Core stabilizing lugs are located on the bottom shell as well as core stop lugs. A flow baffle is located at the bottom head shell course near the core stop lugs. A vent tube through the top head also is present but does not present a fatigue problem. The support of the vessels for all W and CE vessels is at the nozzles, but the usage factors at these support areas are relatively low.

The W design is somewhat similar to CE design and yields similar results.[b] The CRDM housing in the W design also appears to have relatively high fatigue usage. The CRDM nozzles in some cases are made of cast stainless steel that has the potential to degrade because of thermal aging. The same areas of potential degradation turn up regardless of whether a two-, three-, or four-loop plant (i.e., number of nozzles) is considered. Also, no difference between vessel fabricators is seen.

For the B&W design, a list of high usage factor locations for a typical B&W system were obtained.[c] Those regions for the reactor pressure vessel were the closure studs, instrumentation nozzles, and the core flood nozzle venturi sleeve. Interestingly, the inlet and outlet nozzles were not included, nor were the CRDM housings.

Therefore, in general, the regions of greatest concern for fatigue damage are:

1. Closure studs
2. Outlet nozzles
3. Inlet nozzles
4. Instrumentation nozzles (penetrations)
5. CRDM housings/penetrations (nozzles).

Comparing fatigue usage factor values for different regions of a component or calculations between different vendors can be misleading because different simplifying assumptions are made in their computations. The ASME Code, Section III,[1] specifies that the cumulative fatigue usage factor cannot exceed a value of one. Therefore, if a value less than unity is obtained by making simplifying assumptions to allow an easy calculation, there is no need to continue. Refined calculations can significantly reduce the usage factor values reported. These refinements include extensions from simple elastic calculations to simplified elastic-plastic calculations and to highly detailed fully plastic calculations. Other refinements in the way that transients are grouped and counted can also make a significant change.

Corrosion fatigue can occur anywhere fatigue in general is a consideration. Therefore, the sites indicated above are appropriate.

For wear problems, W has identified thimble tubes as an area of concern.[8] Evidently, this concern stems from the fact that some French reactors have experienced wear problems induced by vibration. Replacement appears to be an economically viable option if and when problems arise, although W indicates that there is no significant effect on life extension. The other vendors did not indicate any problems relative to their equivalent of thimble tubes. Thus, wear is not considered to be a significant problem.

a. Private communication with E. Siegel, Combustion Engineering, 1986.

b. Private communication with S. L. Abbott, Westinghouse Electric Corporation, 1986.

c. Private communication with A. D. McKim, Babcock & Wilcox Company, 1986.

3.4 Degradation Mechanisms

The primary degradation mechanisms covered in this section are irradiation damage (embrittlement) and fatigue (both classical design fatigue and crack growth using fracture mechanics). The effects of environmentally assisted fatigue growth also are discussed. The list of potential degradation mechanisms can be very large, because many mechanisms are always present but specific conditions suppress their direct influence. General corrosion and stress-corrosion cracking are not a problem in PWR vessels because the water is kept very clean (and essentially free of oxygen). Erosion and cavitation are not a problem because flow velocities are within acceptable limits. High-temperature creep is not a problem because the temperatures in the vessel are well below one-half the melting point (on an absolute temperatures scale) of the materials. Thermal aging of cast stainless steels is not considered a problem (CRDM housings), because the temperatures are not sufficiently high to promote large changes in properties. The following discussions reflect only those mechanisms that the authors feel to be most significant.

3.4.1 Irradiation Embrittlement. Irradiation-induced increases in yield strength and decreases in the fracture toughness of low alloy steels may limit the operating life of PWR pressure vessels. The major variables controlling such "irradiation embrittlement" are the copper and nickel content of the steels and the total neutron exposure or fluence. Other variables believed to be important include the irradiation temperature, neutron spectrum and flux, phosphorous content, thermomechanical history, concentrations of some other impurity, and minor alloying elements. Indeed, embrittlement may be controlled by the particular combination of these first- and second-order variables; this combination greatly complicates the problem of developing reliable embrittlement predictions based on a limited experimental data base.

Embrittlement is normally monitored by testing samples of base metal, heat-affected zone (HAZ), and weld in the form of CVN and tensile specimens that have been irradiated in surveillance capsules located near or adjacent to a vessel wall. This represents an effort to come as close as possible to simulating the actual conditions experienced by the vessel itself. Federal regulations[9] require that the samples be selected from the most limiting material. There are usually several surveillance capsules which are tested at specified intervals over the life of a vessel; the neutron flux in the surveillance capsules is greater than that in the actual vessel by a lead factor that is intended to provide an early warning of unacceptable levels of property degradation. The CVN specimens provide two measures of embrittlement: changes in the upper-shelf energy (USE) and shifts in the transition temperature at the 30 ft-lb (41 J) energy level, ΔT. The former is used as a signal of significant degradation in the upper-shelf ductile toughness level based on a criterion of the USE falling below 50 ft-lb (68 J). The latter is used to shift RT_{NDT}, the temperature used to reference a lower bound toughness curve, i.e., $\Delta T = \Delta RT_{NDT}$.

Excellent descriptions of most of the vendors' surveillance programs are presented in Reference 10. Certain nuclear power plants do not have any active program because of the structural failure of capsule holders. One approach to dealing with this situation is to use data from other plants, including irradiation of extra capsules in selected host reactors. The use of host reactors is an approach used by B&W after the damage of all surveillance capsule holder tubes in their reactors in operation in 1976.[11] There are two categories of plants having surveillance programs of limited applicability because the materials in the program do not adequately represent the limiting physical properties and chemical compositions of the RPV materials. In the first category of plants, the material in one representative utility surveillance program is actually part of the material from another utility's RPV. In the second category of plants, there may be an undesirable weld material in a number of reactor vessels that is reflected in the surveillance program material of only one of the vessels. The plants in the above two categories may benefit by the industry sponsored programs that provide the unified data from the pre- and postirradiation testing of the specimens.

In general, however, estimates of USE changes and shifts in such circumstances are based on correlations (of the surveillance data base) that are contained in the USNRC Regulatory Guide 1.99.[12] A second revision of this regulatory guide has been proposed and is currently out for public comment.[13] The differences between the two versions of the Regulatory Guide are: copper and nickel concentrations are now taken into account for both weld and base metal, a distinct margin is used to account for uncertainties in the data base, and the damage change through the wall uses a displacements per atom (dpa) approach rather than taking fluence for energies >1 MeV. Regulatory Guide 1.99, Revision 2, provides methods for calculating numerical estimates of changes in USE as a function of copper and fluence, and shifts in transition temperature as a function of copper and nickel content and fluence. Extrapolations of radiation damage into the vessel wall are based on a prototypical fluence and spectral gradient and the assumption that calculated dpa provides an appropriate estimate of spectrum weighted radiation damage exposure. Limitations of the application of the numerical estimates are specified for classes and grades of steels, temperatures, and composition ranges.

The consequences of the new regulatory guide affect at least two areas: the pressurized thermal shock screening criteria and heatup/cooldown curves. With regard to the pressurized thermal shock issue, the higher magnitudes of the new shifts could potentially lead to a reduced residual life in some utility vessels depending upon their exact chemistry and fluence conditions. The lower attenuation through the vessel wall calculated using Regulatory Guide 1.99, Revision 2, than that calculated using Revision 1 means that the heatup curves can now become more limiting.

A major source of uncertainty in current correlations is the limited accuracy and variable range of the surveillance data base. For example, there are little or no data for high-fluence and low-flux combinations that may characterize the end-of-life or life extension conditions of some vessels.

One method of recovering CVN toughness and tensile properties is to perform postirradiation heat treating (annealing). This technique has been successfully applied to two vessels at temperatures less than or equal to 650°F (343°C).[13] For higher temperatures [up to 850°F (455°C)], the feasibility of this technique has been shown analytically[14] but it requires a major engineering effort to actually perform the annealing and sophisticated calculations of the postannealing reembrittlement rate. The complexities in estimating irradiation damage are increased when the effects of an annealing are included.[15] Adjustments in surveillance testing also are required. Even though annealing may be physically possible, the economics have only dictated two attempts: the Army SM-1A vessel in 1967 and the Belgian BR-3 vessel in 1984.[14] Both of these successful annealings were conducted at a lower temperature than is needed for commercial United States reactor pressure vessels. Therefore, more work leading to a demonstration of a higher-temperature annealing for a full-size vessel is needed.

There have been a very large number of accelerated materials test reactor (MTR) embrittlement experiments over the last several decades. It is now generally accepted that because of the high flux levels in these studies, the MTR data should be used as a supplement rather than as a direct substitute for surveillance data. Most of this research has been aimed at generating engineering data (e.g., CVN shifts) for establishing selected effects such as the influence of copper content on shifts or the degree of recovery because of postirradiation annealing treatments.

Sustained efforts to establish the underlying mechanisms of embrittlement have been initiated only recently, primarily by groups in the United States, Great Britain, and Germany. However, substantial improvements in our understanding of embrittlement have been achieved through these efforts.[16-27] Indeed, this research has resulted in significant improvements in the reliability of data correlations. For example, based on the guidance of early theoretical models, correlations of weld surveillance data were developed with standard errors of only ∼16% of the mean shift;[15] more recent analysis has improved the physical basis for the models while maintaining a similar correlation accuracy for an expanded surveillance data base for weldments. The basic insights from the recent studies[17-19] can be summarized as follows:

1. Embrittlement is primarily caused by irradiation-induced increases in the yield strength, which can be quantitatively related to CVN USE changes and temperature shifts.
2. Yield strength increases are because of irradiation-induced, fine-scale microstructures (∼1 nm) that act as obstacles to dislocation motion.
3. The most likely candidate microstructures are precipitates, vacancy clusters (microvoids), and interstitial clusters (dislocation loops). Copper-rich precipitates, possibly alloyed with either elements such as manganese and nickel or vacancies, are believed to be a dominant hardening feature in sensitive steels containing significant concentrations of this impurity element. Other possible types of irradiation-induced or -enhanced precipitates are phosphides and small carbides.

4. The major effects of irradiation are enhanced diffusion rates and defect clustering.
5. These microstructural evolutions are kinetic phenomena and are functions of flux and temperature, as well as composition and microstructure. In general, the changes in the microstructure can be understood from basic principles of alloy thermodynamics and precipitation kinetics, coupled with rate theories of radiation damage.[17-20]

A number of detailed issues remain to be resolved, including: the composition of copper-rich phases and the balance between defect and precipitate microstructures, the mechanisms of nickel enhancement of embrittlement, quantitative treatment of neutron flux effects, the role of factors such as other elements and thermomechanical treatment history, and the kinetics and mechanisms of postirradiation anneal recovery and reembrittlement. Also, the effects of such variables as flux, composition, and annealing conditions on recovery/reembrittlement need further resolution.

In principle, some other mechanisms may contribute to embrittlement in a synergistic, additive, or competitive fashion. These include: thermal aging of ferritic materials caused by precipitation (which is not irradiation enhanced), segregation (temper embrittlement and strain aging), and hydrogen embrittlement/hydrogen attack. For example, there is some evidence that fine-scale copper precipitation can take place at vessel operating temperatures over periods on the order of 25,000 to 250,000 h.[17] Hydrogen effects could be promoted by circumstances that may or may not be present in PWR environments, related to nonequilibrium partial pressures or local chemical reactions that yield high local hydrogen fugacities.

There are a number of other unresolved questions. For example, the accuracy of using CVN-based RT_{NDT} shifts to reference toughness curves (ASME Code Section III, Appendix G) has not been demonstrated. Theoretical considerations suggest that in some sensitive steels this practice may be nonconservative, and that relatively low cleavage fracture toughness levels may persist up to elevated temperatures in steels subject to very large irradiation-induced yield strength increases.[16] In other words, the actual RT_{NDT} shifts in some pressure vessel steels may be significantly larger than the CVN-based RT_{NDT} shifts. Indeed, the seriousness of this issue may be mitigated only by the uncertain conservatism of using RT_{NDT}-referenced lower bound toughness curves. (It should be noted that some work is being pursued within the Heavy Section Steel Technology (HSST) Program sponsored by NRC to actually measure shift and shape change in the K_{IC} and K_{Ia} curves from large-specimen fracture toughness tests).[28] Further, no reliable basis exists for characterizing irradiation effects on ductile upper-shelf static fracture toughness and crack arrest toughness from surveillance data. All these properties may be required in a comprehensive integrity analysis.

Finally, this section has not covered the topics of defect monitoring and evaluation or application of the material properties in fracture mechanics analyses, including the treatment of elastic-plastic effects and multidimensional part-through flaws subject to combinations of primary and secondary stresses (which may be rapidly time varying). Information on the effects of irradiation on additional properties of ferritic steel, such as fatigue crack growth rates (including the effect of the environment), stress corrosion cracking, strain rate effects, strain hardening exponents, and various measures of ductility are generally required to carry out such analysis. Indeed the effect of radiation damage on the overall constitutive behavior and macroscopic/microscopic strain distribution may be important. It is noted that if large increases in yield strength are correlated with large decreases in cleavage fracture toughness, a greatly expanded regime of brittle vessel operation might be anticipated.[17]

3.4.1.1 Uncertainties. The prior discussion sets the context for uncertainties in characterizing the material state as a function of service life and history for use in analysis of pressure vessel safety margins. Summarizing, uncertainties are related to:

1. The accuracy of either plant specific or correlation based estimates of CVN transition temperature shifts for specified values of the dominant variables
2. The existing surveillance programs do not include the effects of high-fluence and low-flux conditions, and treat the effects of spectrum and irradiation temperature approximately.
3. The actual versus nominal value of significant variables in the most critical location in the vessel
4. The questionable validity of using reference toughness curves and shifting these

curves by transition temperature increases measured in CVN tests

5. The essential lack of information on a number of properties, such as static upper-shelf toughness (which may be needed in an integrity analysis in general); and an even greater lack of information on the effects of in-service irradiation on these properties. In addition, there are uncertainties in the initial properties such as the unirradiated RT_{NDT}.
6. Gaps in basic understanding and detailed empirical characterization of many aspects of embrittlement, postirradiation annealing, and reembrittlement phenomena.

While this admittedly incomplete list of technical uncertainties is imposing, it may not be all that significant. Currently, it is simply not possible to assess the combined effect of the issues listed above on the general reliability of the integrity assessments. However, a change in attitude about how to treat the aggregated uncertainties, or discovery of new issues (e.g., an unanticipated accident sequence) could change the assessment of residual life far more drastically than would be expected from further resolution of some technical issues listed above (e.g., the accumulation of additional CVN surveillance data or the identification of a new second-order variable or embrittlement mechanism). Clearly, the magnitudes of the technical uncertainties increase with the length of the projected time frame; and may be paramount to the issues of relicensing and life extension.

3.4.1.2 Practical Issues. There are a number of practical issues that constrain the range of approaches for resolving the uncertainties indicated above. Some of the technical issues are being addressed by the USNRC research[29] and others, but it is not totally clear that all issues are integrated relative to life extension considerations. Some of the pertinent practical technical issues include:

1. The limited range of potentially important variables and properties contained in the existing data base and the reliability of subsets of that data base.
2. The limited amounts of critical material and limited numbers of surveillance capsules. (For example, if plant life extension is anticipated, how should this influence the schedule and approach to testing surveillance capsules? Or, what can be done after all the capsules are used up?)
3. The limited amount of property information that can be obtained from current CVN-based surveillance programs relative to the data needed for comprehensive integrity assessments.
4. The flexibility (or inflexibility) of existing regulation and ASME code requirements.
5. Questions regarding the suitability of the use of MTR irradiations to simulate radiation damage under service conditions; and the validity of using (prototypical) surrogate alloys to represent the behavior of vessels materials (e.g., if archival samples are not available).
6. Lack of a pertinent data base to assess the potential for a long-time, irradiation-induced aging effects in ferritic steel.

3.4.1.3 Issues to be Resolved. Clearly, the most pressing need is the development and timely execution of a comprehensive, integrated, and balanced plan for dealing with the array of issues associated with residual life assessment and life extension of reactor pressure vessels. The NRC Materials Branch has such a plan underway for the reactor pressure vessel, i.e., the HSST Program.[30] Timely execution of this plan will allow resolution of several issues associated with aging of the reactor pressure vessels. It is not appropriate to attempt to design such a plan here; however, it is useful to elaborate on the approach and to identify a few key elements that can be expected to emerge from a detailed study:

1. Comprehensive—After identifying the key variables, develop a means of obtaining and validating the necessary array of properties for alloys and conditions (i.e., the irradiation, thermal and chemical environment) reasonably representative of the realities of long-term in-service behavior.
2. Integrated—The experimental effort to obtain appropriate properties in suitably conditioned alloys should be closely tied to the overall basis for integrity analysis. The integration with the mechanics component of the integrity analysis should probably anticipate significant advances in computational capabilities. On one hand, such development of computational capabilities may permit the further exploitation of

miniature specimen testing and reconstituted specimen testing techniques (see Chapter 14 for additional discussion), greatly extending the potential sources of information about embrittlement; and, on the other hand, these advances will permit more direct and reliable application of fundamental property data to practical structural analysis. Significant improvements in the basic theoretical understanding of the mechanisms of embrittlement and techniques to characterize the embrittling microstructures (as well as property changes) should also be anticipated.

3. Balanced—The efforts should be carefully balanced, because the impact of a few weak links in the sequence of analysis will far outweigh high precision in one or another step in the process. For example, it would do little good to develop methods to account for each and every neutron impinging on a surveillance capsule or vessel if the irradiation temperature or alloy microstructure and composition were still poorly characterized.

Some of the important specific elements in this program are listed below (the HSST program has some activities planned in many of these elements):

1. Rapidly resolve critical outstanding issues, e.g., the potential for high-temperature cleavage fracture in sensitive steels with high levels of irradiation strengthening
2. Comprehensively define the property array needed to carry out reliable integrity analysis, including ranking of the overall sensitivity to uncertainties in the properties
3. Develop improved ways of extracting an array of mechanical property information from material in surveillance programs, e.g. miniature specimen techniques
4. Validate available RT_{NDT} shift data from accelerated MTR irradiations as a basis for establishing long-term in-service behavior for RPV steels. These data will supplement the surveillance data from the power reactors
5. Hold additional surveillance specimens in power reactor internals above or below the core as well as on the sides to get the appropriate flux and fluence levels, and temperatures
6. Initiate some long-time studies of other environmental effects (e.g. thermal aging), and their potential synergistic interactions with irradiation embrittlement
7. Resolve postirradiation annealing recovery/reembrittlement kinetics and mechanisms
8. Optimize surveillance programs with long-term objectives clearly defined
9. Continue efforts to develop fundamental understanding of embrittlement and postirradiation annealing recovery/reembrittlement phenomena in terms of physically based models and techniques to characterize the state of the alloy (such as small-angle neutron scattering).

3.4.2 Fatigue. The fatigue analysis of Class 1 components in nuclear plants is defined by the ASME Boiler and Pressure Vessel Code, specifically in Section III for the fatigue design evaluation procedure[1] and in Section XI for the flaw acceptance standards and evaluation procedures.[31] The Section III procedures are based on the classical stress-strain/life (S/N) approach, using an experimentally determined relationship between the elastically calculated stress range and fatigue life. Under variable amplitude cyclic loading, a Miner's rule approach of accounting for damage accumulation in the component at different stress ranges is used by linearly summing the fraction of life consumed at each stress range with the cumulative usage factor set at less than, or equal to, unity. Section XI procedures, conversely, rely on fracture-mechanics-based, damage-tolerant analyses, where continued operation of components containing defects can be permitted in-service, provided predictions of fatigue-crack growth show that the defect will remain subcritical until the next inspection.

From the perspective of improved life-prediction procedures, the Section III and XI approaches may be open to question. First, they represent fundamentally different fatigue approaches (one based primarily on crack initiation and the other based on crack growth), which may cause problems of consistency when applied to the same component.[32] Second, with regard to the S/N analysis, no allowance is made for the sequence of loading cycles with the Miner's linear damage law, and further separate analyses are required for high- and low-cycle fatigue. Third, the fatigue-crack-growth, damage-tolerant analysis does not fully account for such factors as near-threshold behavior, variable amplitude loading, and the anomalous behavior of small flaws (typically of a size < 1 mm).[33]

Described next are various considerations that may be used to improve the life-prediction capability of the Section III and XI fatigue design analyses. (Note that the NRC Pressure Vessel Research Program is conducting work on both environmental fatigue-crack-growth behavior and cumulative fatigue-damage modeling.) Such advanced lifetime computation procedures are designed for life extension and rely on:

1. A modified S/N procedure for crack initiation based on the local-strain approach,[34] which incorporates the effects of both stress concentrations at notches and the sequence of variable amplitude cycles
2. An updated version of the damage-tolerant procedures that incorporate an improved appreciation of environmental factors, low growth rate behavior, and the role of small flaws.

3.4.2.1 Modified Life Prediction Approach for Crack Initiation. The local strain-approach represents an advanced version of the classical S/N approach, and provides a more consistent approach to fatigue life prediction, based primarily on crack initiation. Specifically, it incorporates improved analyses of the roles of mean stress and stress concentrations and in the summation of cumulative damage.[34] The analysis is essentially based on total strain, and so as such, does not require any distinction between high- and low-cycle fatigue. Typical S/N analyses generally define low-cycle fatigue (LCF) as failure in the range of 10^2-10^5 cycles (depending upon the material) at stresses near or above yield, and high-cycle fatigue (HCF) as failure at a high number of cycles under nominally elastic stresses. The Coffin-Manson law can be used for life prediction in LCF, and the Basquin relationship can be used in HCF.[35]

The first commonly used equation for the local-strain approach is the experimentally-determined cyclic stress-strain curve, which relates cyclic stress range ($\Delta\sigma$) to strain range ($\Delta\epsilon$) in terms of Young's modulus (E) and the cyclic strain hardening exponent (n'), and is written

$$\frac{\Delta\epsilon}{2} = \frac{\Delta\sigma}{2E} + \left(\frac{\Delta\sigma}{2k'}\right)^{1/n'} \qquad (3.1)$$

where k' is the cyclic-strength coefficient representing the slope of the log stress/log plastic strain curve, and n' is typically 0.15. Use of this flow rule allows a far better estimate of actual material behavior (e.g., cyclic hardening or softening). The second commonly used equation is the basic stress-strain/life equation, which relates total strain range ($\Delta\epsilon$) to life (N_f) by incorporating modified versions of the Coffin-Manson law for LCF, the Basquin relationship for HCF, and the Goodman relationship for mean stress (σ_0) into a single equation that gives[34,35]

$$\frac{\Delta\epsilon}{2} = \frac{\sigma_f' - \sigma_o}{E}\left(2N_f\right)^b + \epsilon_f'\left(2N_f\right)^c \qquad (3.2)$$

where σ_f and ϵ_f are the fatigue-strength and fatigue-ductility coefficients, respectively (approximately equal to the true fracture strength and fracture strain in a tensile test), and b and c are the fatigue-strength and fatigue ductility coefficients, respectively (with approximate values of -0.1 and -0.6). It should be noted here that no fatigue limit is assumed.

Equation (3.2) is also used for the prediction of fatigue life in the presence of stress concentrations, where the values of $\Delta\sigma$ and ΔE now define the local stresses and strains at the notch root. Instead of employing the classical approach of computing these local stresses merely in terms of the theoretical stress concentration factor (K_t) or the fatigue notch strength reduction factor (K_f), Neuber's rule,[36] or equivalent numerical calculations, can be used to compute the local strain history. For example, Neuber's rule gives

$$K_f = (K_\sigma K_\epsilon)^{1/2} \simeq \frac{(\Delta\sigma\,\Delta\epsilon\,E)^{1/2}}{\Delta S} \qquad (3.3)$$

where K_σ is the actual stress concentration (ratio of local to nominal stress, $\Delta\sigma/\Delta S$) and K_ϵ is the actual strain concentration (ratio of local to nominal total strain, $\Delta\epsilon/\Delta e$). Such an approach can result in much improved life prediction, particularly in lower strength steels where local yielding at the notch results in significant strain concentration as well as stress concentration.

For irregular cyclic loading conditions, a modified Palmgren-Miner linear-cumulative-damage rule can be employed to allow for consideration of the sequence of variable amplitude loading cycles. (Sequence effects, overstressing, and understressing are not accounted for in classical S/N analyses.) The basic linear-cumulative-damage law can be stated as

$$\sum_i \frac{n_i}{N_{fi}} = \text{usage factor} \tag{3.4}$$

where n_i is the number of cycles of a given stress range ($\Delta\sigma_i$), N_{fi} is the number of cycles to failure had all the cycles been of magnitude ($\Delta\sigma_i$), and the usage factor is set not to exceed unity. In the modified approach, which has been shown to yield superior fatigue-life estimates with complex loading histories,[37] the primary input is to account for mean stress and sequence effects by using a more appropriate means for cycle counting. There are several procedures that have been used with success, but the so-called rainflow method is generally the most widely used.[37] This method involves:

1. The determination of the number of complete hysteresis loops in the loading spectrum
2. Measurement of a mean and alternating stress (or strain) for each loop
3. Computation of the value of N_f for each loop from Equation (3.2)
4. Calculation of the usage factor, using Equation (3.4) in the usual manner.

Because these procedures require a great deal of tedious bookkeeping to implement, several algorithms have been published to reduce computation time.[38]

Finally, as with all fatigue analyses, it is vital to consider the effect of service environment, as it is well known that fatigue life can be drastically altered by the presence of an active environment (such as PWR water). This generally entails defining the life equation [Equation (3.2)] in simulated service environment with loading applied at a realistic frequency, if possible.

Compared to classical S/N analyses, the local-strain approach has two principal advantages. First, life predictions for a wide variety of situations can be made from a limited amount of small-specimen test data. Second, it provides a rational basis for considering various material behavior-related interaction effects that are important in metal fatigue. For these reasons, the approach yields superior life predictions compared with classical analyses, and could be successfully used to better define usage factors for life-extension considerations.

3.4.2.2 Modified Life Prediction Approach for Crack Propagation. The flaw-evaluation procedures in Appendix A of Section XI of the ASME Code[31] use linear elastic fracture mechanics to predict the growth of existing flaws under cyclic loading. For these procedures to be effective, crack extension rates (da/dN) must be predicted on the basis of a realistic crack growth relationship, which generally has the form[39]

$$\frac{da}{dN} = f(\Delta K, K_{max}, F, \ldots) \simeq C(\Delta K)^m \tag{3.5}$$

where K_{max} and ΔK are the maximum value and range of the stress intensity factor ($K \cong \sigma\sqrt{\pi a}$), F is the frequency, and C and m are experimentally determined constants appropriate to the material the frequency, and C and m are experimentally determined constants appropriate to the material intermediate cyclic crack growth rates and can be used with altered (realistic) constants for some portions outside the intermediate crack growth range. The approach of James and Jones[40] for austenitic stainless steels represents an ideal example. These authors employed an extensive data base for stainless steels in order to apply least squares regression techniques to derive a power-law, crack-growth relationship [Equation (3.5)] for use in life calculations. However, because in this particular case no attempt was made to allow for environmental effects, the equation can only be used for crack growth behavior in air.

The approach of integrating an appropriate crack growth relationship from initial to final defect size[41] is generally considered to be inherently conservative because it assumes negligible initiation life. However, based on recent studies of variable-amplitude, near-threshold, and small-crack behavior (e.g., References 33, 42, 43), there are several factors that should be considered additionally to improve the accuracy of life prediction. These factors are discussed later.

3.4.2.3 Data Base Establishment. Crack propagation data should be generated experimentally in as realistic environments and frequencies as possible, over a wide spectrum of growth rates. Sufficient data must be obtained to guarantee confidence in the resulting equations. Data in the ultra-low growth-rate regime below 10^{-6} mm/cycle, approaching the so-called fatigue threshold, are particularly important as fatigue lifetimes, especially those of practical importance to life extension, can be dominated by crack extension in this region. For example, using a linear extrapolation of higher growth rate data (above 10^{-5} mm/cycle[41] may lead to heavy penalties on predicted life and indicate component retirement long before its safe useful end-life is reached). The need for ultralow growth rate data is further amplified by estimates of reactor component service loads. For example, service loads at the Zion-1 pressurized water reactor plant have been estimated to include over 2 x 10^{10} cycles of vibrational stress cycles [of magnitude typically 1000 psi (6.89 MPa)],[32] which over the 40-year life of a particular component could lead to extensive crack growth at the seemingly insignificant near-threshold growth rates. Because near-threshold growth rates can be severely retarded by the phenomenon of crack closure[42] in laboratory tests (e.g., using compact-type specimens) at low load ratios, it is essential that such data be determined at high load ratios above 0.5 to generate a conservative crack-growth relationship (see also small crack discussed later).

3.4.2.4 Variable Amplitude Behavior. Unlike crack initiation under irregular cyclic loading, as described by Miner's law [Equation (3.4)], crack growth under variable amplitude loads can be subjected to severe load interaction effects, particularly for long cracks under broad band spectrum loading.[44] Such interaction effects generally take the form of transient crack-growth retardations, following, say, single tensile overload cycles or high-low block loading sequences, which contribute significantly to enhanced fatigue lifetime.[45] Although physical explanations for such behavior are still mechanistically uncertain, the aerospace industry (in particular) has been successful in devising several modified damage-tolerant analyses that can account (albeit empirically) for the increased life caused by such transient behavior. The most widely used models are those of Wheeler[46] and Willenborg et al.,[47] although more sophisticated approaches based on numerical modeling of crack closure have recently been developed.[48] The approach in all these models is essentially to use constant amplitude data to predict variable amplitude behavior through modification of the integration of the crack growth relationship, e.g., through determining a retardation factor on crack growth while the crack is propagating within the plastic zone of the overload cycle. Incorporation of such procedures in the life prediction analyses may well contribute to increased life (which corresponds essentially to decreased usage factors).

3.4.2.5 Small-Crack Effects. One further factor, which should be considered with advanced damage-tolerant life prediction procedures, is an appreciation of the possible anomalous behavior of small flaws. Cracks are considered small when they are of a size comparable with the scale of the microstructure, or the scale of the local plastic zone size, or when they are simply physically small (i.e., < 1 mm).[33] In general, their behavior is to display growth rates in excess of those of conventional long cracks (e.g., over 10 mm in length), when exposed to the same stress intensity range (although mild reverse effects have been reported). In addition, small crack growth can occur below the fatigue threshold, below which long cracks are presumed dormant. Moreover, under severe environmental conditions, the accelerated growth behavior of small cracks may be further enhanced.[49] Short of generating experimental data on such small cracks, damage-tolerant procedures to account for such behavior have yet to be developed. However, since mechanistically the behavior of small flaws is primarily influenced by a diminished role of crack closure, by generating long crack data at high load ratios, some of the discrepancy between long- and short-crack growth rates can be diminished.

In summary, it is felt that by incorporating advanced S/N analyses to replace the typical approaches outlined in Section III of the code, and by incorporating revised damage-tolerant, crack-growth procedures to supplement the Section XI procedures, significantly improved life-prediction capabilities can be obtained. Such approaches could further be embellished by consideration of the small-crack issue, which represents the interface between crack initiation and crack propagation analyses.

3.5 Potential Failure Modes

The eventual failure modes resulting from degradation during service life will generally emerge when a severe abnormal transient or event occurs. For example, in the reactor beltline region, if a pressurized thermal shock event were to develop and the material properties were already severely degraded by irradiation embrittlement (i.e., the RT_{NDT} was raised to high temperatures), the potential for failure of the vessel in a brittle manner is possible. However, there must also be a defect or crack of critical size present to produce this failure scenario. The possibility of an irradiation-induced low upper-shelf material in the beltline region could lead to a low-energy ductile tearing overload situation, if again a crack is present. Therefore in the beltline region, the necessity of having a defect is paramount for producing failure. However, the regions of interest in the beltline have extremely low fatigue usage factors indicating the small probability of having a growing defect or crack. Defects are possible as evidenced by the concern over underclad cracks in the 1970s;[50,51] these manufacturing-induced cracks can result from high heat input multipass strip cladding techniques (at the overlap points), especially for SA508-2 forging steel. The United States processes typically use a single- or double-layer submerged arc pass, and there is no embrittlement of the HAZ, leading to cracks. The United States vessels have not shown in-service evidence of underclad cracking. These small cracks, even if present, would not pose an integrity problem because a series of them would have to link together to form a continuous defect of any significant size.[6]

In the areas where fatigue is a potential problem, the type of final failure will almost always result from a plastic overload situation caused by the stress concentration and reduced ligament load carrying ability (i.e., the resultant material left from the fatigue crack tip to the surface.) These ductile tearing conditions are not necessarily failures, but are potential through-wall leakers. However, the inlet nozzles could be exposed to cold water during some accident conditions that could lead to a brittle fracture there (especially if some irradiation damage also was present). The CRDM housings and instrumentation penetrations would probably be exposed to only ductile overload situations, and the closure studs also would be exposed to only overload conditions resulting in ductile failure for conditions caused by internal pressure, boltup loads, and some thermal transients.

In planning for extended life, the closure studs should be replaced near the end of the design life of the vessel because of the high fatigue usage. The CRDM housings, instrumentation penetrations, and outlet/inlet nozzles should be monitored closely and will most likely limit the vessel extended service life before repair or replacement is required. The beltline region may be most limiting, but each individual vessel will require independent assessment depending on material chemistries, weld types, and fluence conditions.

3.6 In-service Inspection and Surveillance Methods

In-service inspection (ISI) is required by the ASME Code, Section XI.[31] The general philosophy of Section XI ISI and the status of the improvements in the code requirements for the RPV are presented in Chapter 12. There are basically four inspection intervals, and certain welds must be examined during those intervals. All shell, head, shell-to-flange, head-to-flange, and repair welds must be subjected to a 100% volumetric examination during the first inspection interval (over 3 to 10 years). Successive inspection intervals require fewer beltline region, head, and repair weld examinations. The nozzle-to-vessel welds must all be subjected to a volumetric examination during all four inspection intervals. Twenty-five percent of the partial-penetration nozzle welds (CRDM and instrumentation) are required to have a visual, external surface examination during each inspection period (leading to total coverage of all nozzles). All of the nozzle-to-safe end butt welds with dissimilar metals (i.e. ferritic steel nozzle to stainless steel or Inconel) must be subjected to volumetric and surface examinations at each interval. These welds are covered in Chapter 5 (primary system piping) of this report. All studs and threaded stud holes in the closure head studs need surface and volumetric examinations at each inspection interval. Any integrally welded attachments are required to have surface (or volumetric) inspections of welds at each inspection interval. A brief description of the currently used volumetric examination methods is given in Chapter 11.

Thus, the inspection plan is very complete and results in close monitoring of potential fatigue-crack formation and growth. The additional monitoring and recording of transients are usually done in accordance with the plant technical specifications; however, this area could be greatly enhanced by more accurate monitoring and assessment of the transients (identified earlier) leading to fatigue usage.

The surveillance program for monitoring changes in the vessel wall and weld properties caused by neutron embrittlement has been discussed in Section 3.4. Coupling these surveillance results (and existing data) with the accurate characterization of flaws or defects through ISI would allow prudent calculation of vessel integrity. If better logging of the critical fatigue transients also is undertaken, even further confidence in the reliability of the pressure vessel is possible. More confidence now in the safety of PWR vessels will extend directly into later life extension.

3.7 Summary, Conclusions, and Recommendations

A summary of the important degradation sites, mechanisms, and consequences for PWR vessels is shown in Table 3.2. Note that the ranking of sites is based on the importance of that area, not on the possibility of that site having severe degradation. For example, the closure studs are probably the most likely site to have significant damage (except for possibly the reactor beltline region), but the studs are listed at the end because they are easily replaced. Other conclusions and recommendations are:

1. Radiation embrittlement is the primary safety concern that may limit vessel life extension; flux reduction programs used to reduce fluence levels need to be coordinated with overall life extension plans; Charpy referencing of toughness needs to be resolved, fundamental studies of irradiation damage and further work on phenomenological correlation models should be developed for both transition temperature shift and upper shelf energy drop (rate effects need to be better understood and correction factors developed); surveillance program schedules may need adjustment and revision to meet life-extension goals.
2. Thermal annealing needs to be pursued as a contingency plan (a long-term option); annealing mechanisms and kinetics need to be better understood and models developed.
3. Application of miniature specimen testing and reconstituted specimen testing need to be evaluated. Use of these testing techniques is valuable because the available space to irradiate the test specimens and the available test material are limited. This is discussed further in Chapter 14.
4. Other synergisms associated with irradiation may be present and need to be studied along with some simple long-time thermal aging studies; the total package of information needed for structural integrity analysis should be reviewed and uncertainties quantified.
5. Nozzles and penetrations represent the major fatigue problems associated with life extension; better monitoring of fatigue transients is needed to properly assess actual usage factors; more sophisticated fatigue usage factors can be developed using the local-strain approach presented earlier; as actual usage factors become large (i.e., approach unity), fatigue-crack propagation calculations using the state of the art should be used to predict usable life.
6. Flange closure studs should be replaced when considering life extension; also, the thimble tubes in W plants should be inspected and assessed for replacement.
7. Inspection and transient logging records as well as all material properties (unirradiated and surveillance results), design data, and other key information, should be maintained for future reference; early collection of these records will allow life extension plans to be developed.
8. It is probable that the reactor pressure vessel, in most cases, can be used for longer than the 40-year license period dictated by federal regulations.[9] Actual decisions are plant specific, however, due to slightly different designs, different limiting materials, and operating conditions (fluence levels and transient history). Continued close monitoring of the material condition in the vessels is required for life extension.

Table 3.2. Summary of degradation processes for PWR reactor pressure vessels

Rank of Degradation Site	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Surveillance Methods
1	Beltline region	Neutron irradiation, mechanical and thermal stresses	Irradiation embrittlement (degree is dependent on individual vessel materials and flux spectrum history)	Ductile high-energy tearing leading to leakage (net section over-load)	100% volumetric during first inspection; one weld for subsequent inspection
				Brittle fracture (i.e., pressurized thermal shock)	Surveillance program for assessing irradiation damage is required by law
				Ductile low-energy tearing (low upper-shelf toughness)	
			Environmental fatigue	Ductile overload leading to a leak; possible brittle fracture if PTS occurs	
2	Outlet/inlet nozzles[a]	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload leading to a leak; possible brittle fracture if pressurized thermal shock occurs with some irradiation embrittlement	All nozzle welds inspected volumetrically at each interval
3	Instrumentation nozzles (penetrations) and control rod drive mechanism housing nozzles	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload leading to a leak	Visual, inspection of external surface; 25% of nozzles inspected at first interval; remaining 75% spread out over next three intervals
4	Flange closure studs	Mechanical and thermal stresses	Fatigue crack initiation and propagation (possibly corrosion assisted)	Ductile overload failure (can be replaced)	Volumetric and surface inspection of all studs and threads in flange stud holes at each interval

a. Welds at the vessel to safe end are covered in Chapter 5 on primary system piping; the comments here are also applicable except that volumetric and surface examinations are required.

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4. PRESSURIZED WATER REACTOR CONTAINMENTS AND BASEMATS

M. A. Daye

4.1 Description of Containment Structures

The majority of the pressurized water reactor (PWR) containments in the United States are prestressed concrete containment vessels (PCCV) with steel liners covering the inside surfaces. Reinforced concrete containment vessels (RCCV) with steel liners are also commonly used. Other types of containments constructed as steel cylinders or steel spheres were used for some PWRs.

The function of the containment structure is to house the nuclear steam supply system (NSSS) and limit the release of radioactive material to the environment during normal operation and in the event of accidents. In particular, the containment structure provides the much needed shielding against radiation in the event of an accident so that necessary emergency measures can be provided to minimize the consequences and deal with the causes in a protected environment. The structure also provides protection to the reactor system from the effects of severe and extreme environmental loads such as tornadoes, external accidents, and penetration by moving objects.

Containment structures are designed to withstand design earthquakes, loss-of-coolant accident pressures and temperatures, and operating loads and temperatures for a continued, predesignated life span of normally 40 years.

4.1.1 Prestressed Concrete Containment Vessels. The principal components of a PCCV are reinforced concrete, steel liner plate, and posttensioning wire or strand tendons.

In general, the containment structure consists of a flat reinforced concrete slab, an upright posttensioned concrete cylinder and posttensioned concrete dome. All internal surfaces are lined with steel plates anchored to the concrete. Figures 4.1(a) and 4.1(b) represent two configurations for the PCCV containments.

4.1.2 Reinforced Concrete Containment Vessels. The RCCVs are basically similar to the prestressed concrete containment vessels except that the cylinder and dome are not posttensioned and rely completely on the reinforcing steel and concrete for load resistance (Figure 4.2).

4.1.3 Steel Cylinder/Steel Sphere Containment Vessels. Steel cylinder containments consist mainly of a freestanding steel cylinder with a steel dome on top and an ellipsoidal steel bottom that is anchored to a concrete foundation. The inside of the steel bottom is filled with concrete to approximately the transition line between the ellipsoidal bottom and the base of the cylinder. Within the bottom concrete inside the containment, the reactor cavity is formed to house and support the reactor vessel. The freestanding cylinder and dome are enclosed inside a reinforced concrete reactor building that forms the biological shield around the containment (Figure 4.3). The PWR containments with ice condensers, i.e., Sequoyah-1 and -2, and Watts Bar-1 and -2, have this type of containment. Some of the other plants having this type of containment are Indian Point-1, St. Lucie-1 and -2, Prairie Island-1 and -2, and Davis-Besse-1.

The steel spherical containment structure is similar in concept to the freestanding steel cylinder except that the cylindrical portion is omitted and a complete sphere is used. The bottom of the sphere is anchored to a concrete foundation. The inside is filled with concrete to a height sufficient to provide structural stability and form the reactor cavity within. There is one commercial PWR plant that has this type of containment structure in the United States, San Onofre-1. Yankee-Rowe, a 175-MWe demonstration plant, also has a steel spherical containment and it is supported by concrete/steel columns through the lower hemisphere.

4.1.4 General Discussion. Exterior and internal walls and floors of the containment structure are invariably of thick concrete primarily for shielding purposes. Occasionally, high-density aggregate such as ilmenite, is used to provide additional shielding without increasing the thickness of the concrete members.

The base slabs for both PCCV and RCCV containment structures are essentially the same and constructed of reinforced concrete. Some variations in the geometry of the base slabs may exist depending on the NSSS system used. The slab is either completely flat or has a discontinuity where the reactor vessel is located in a cylindrical reactor cavity compartment formed into the slab.

Posttensioning systems consist of high-strength steel wire or strand tendons, end anchorages, and corrosion protection filler material. The tendons

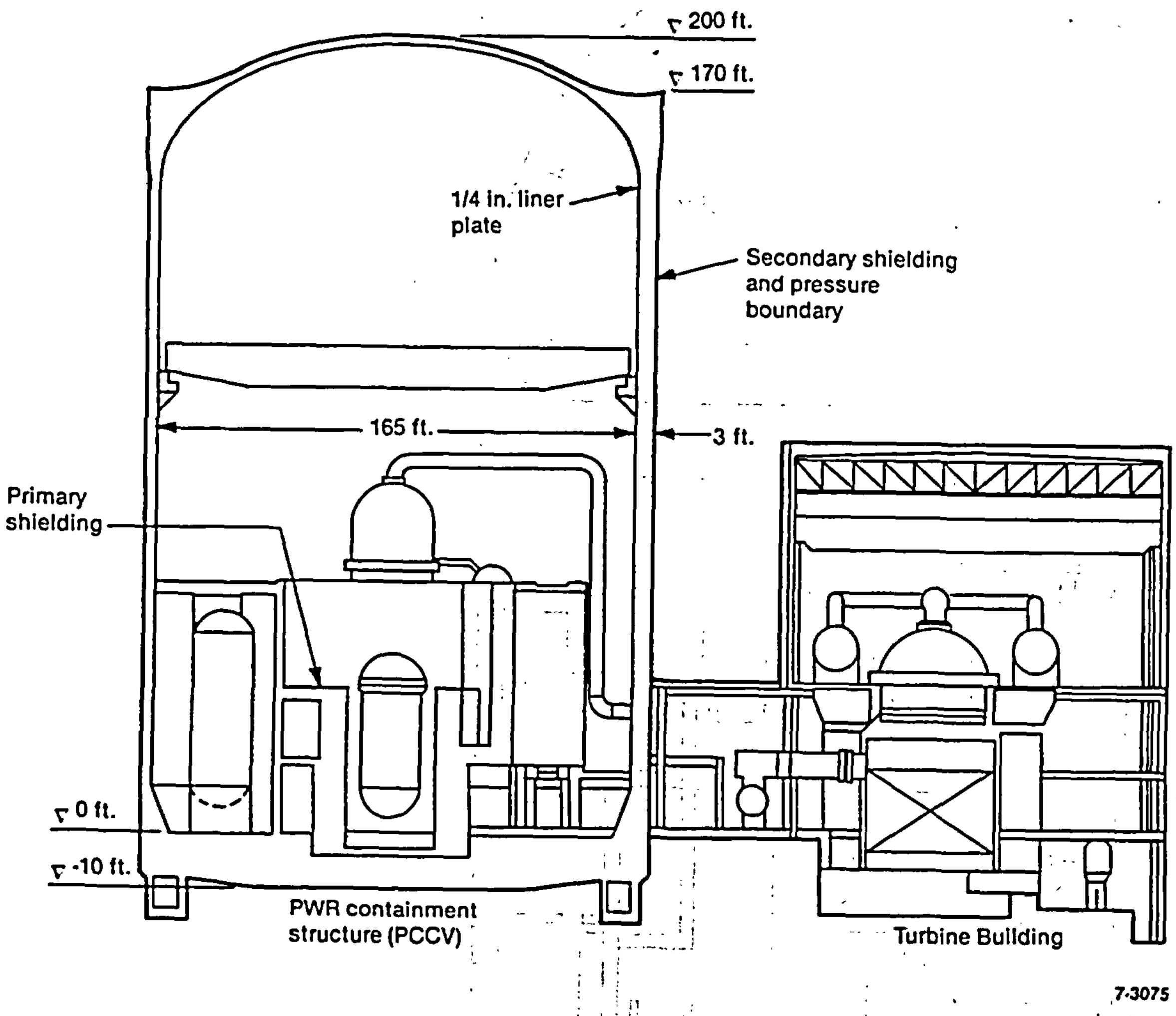

Figure 4.1(a). PWR prestressed concrete containment Type I and Type II.

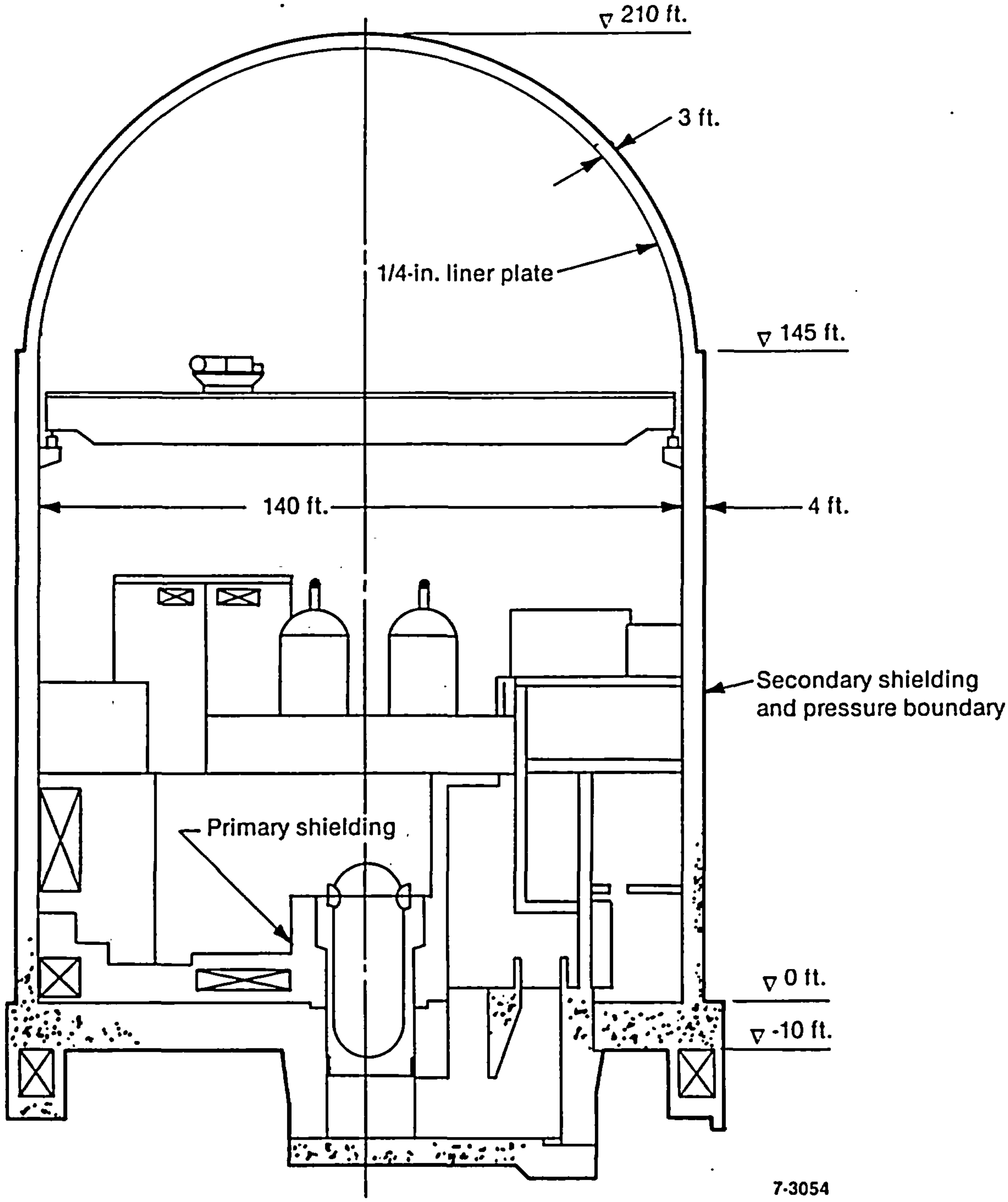

Figure 4.1(b). PWR prestressed concrete containment Type III—three buttresses and hemispherical dome.

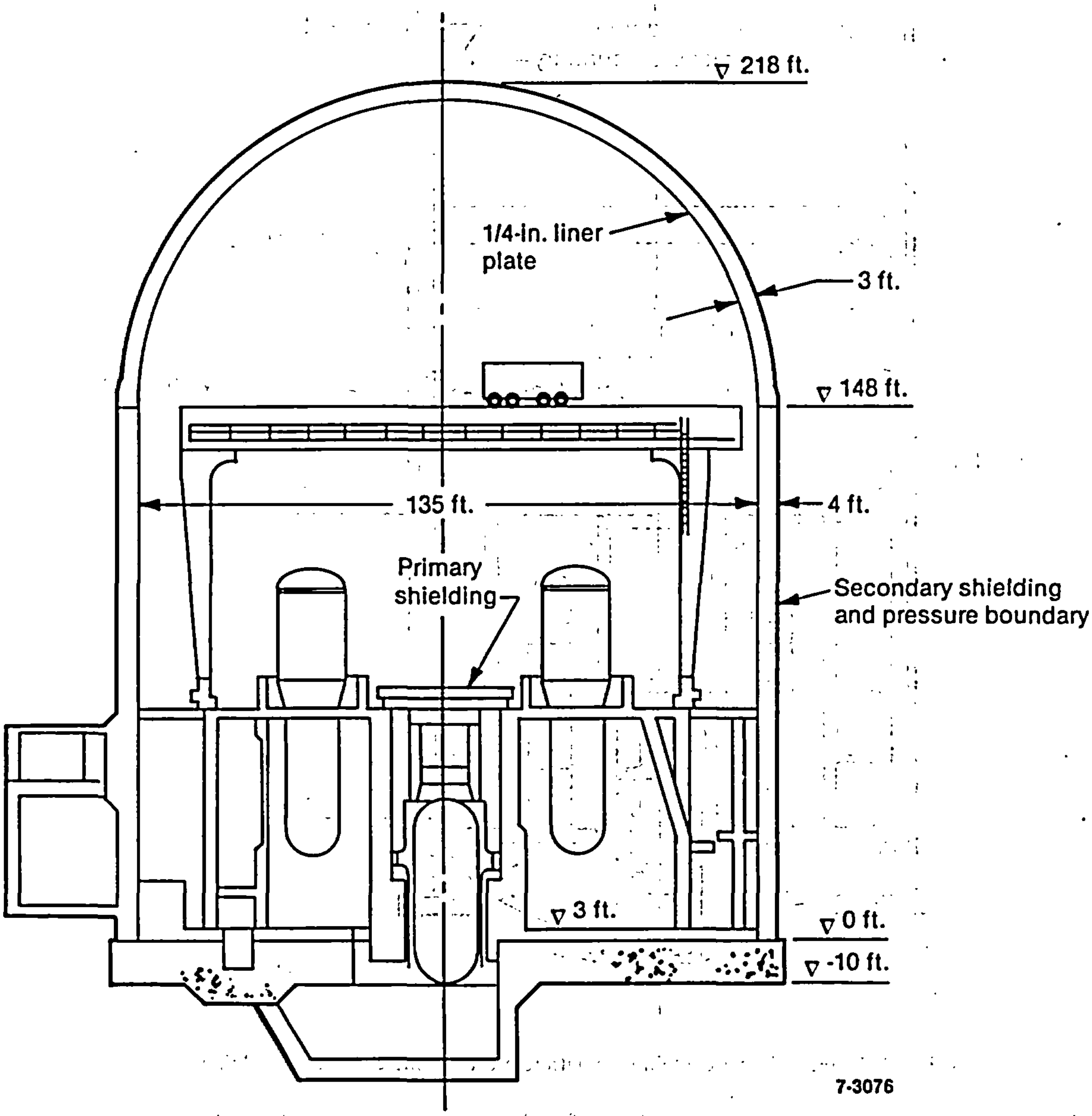

Figure 4.2. PWR reinforced concrete containment.

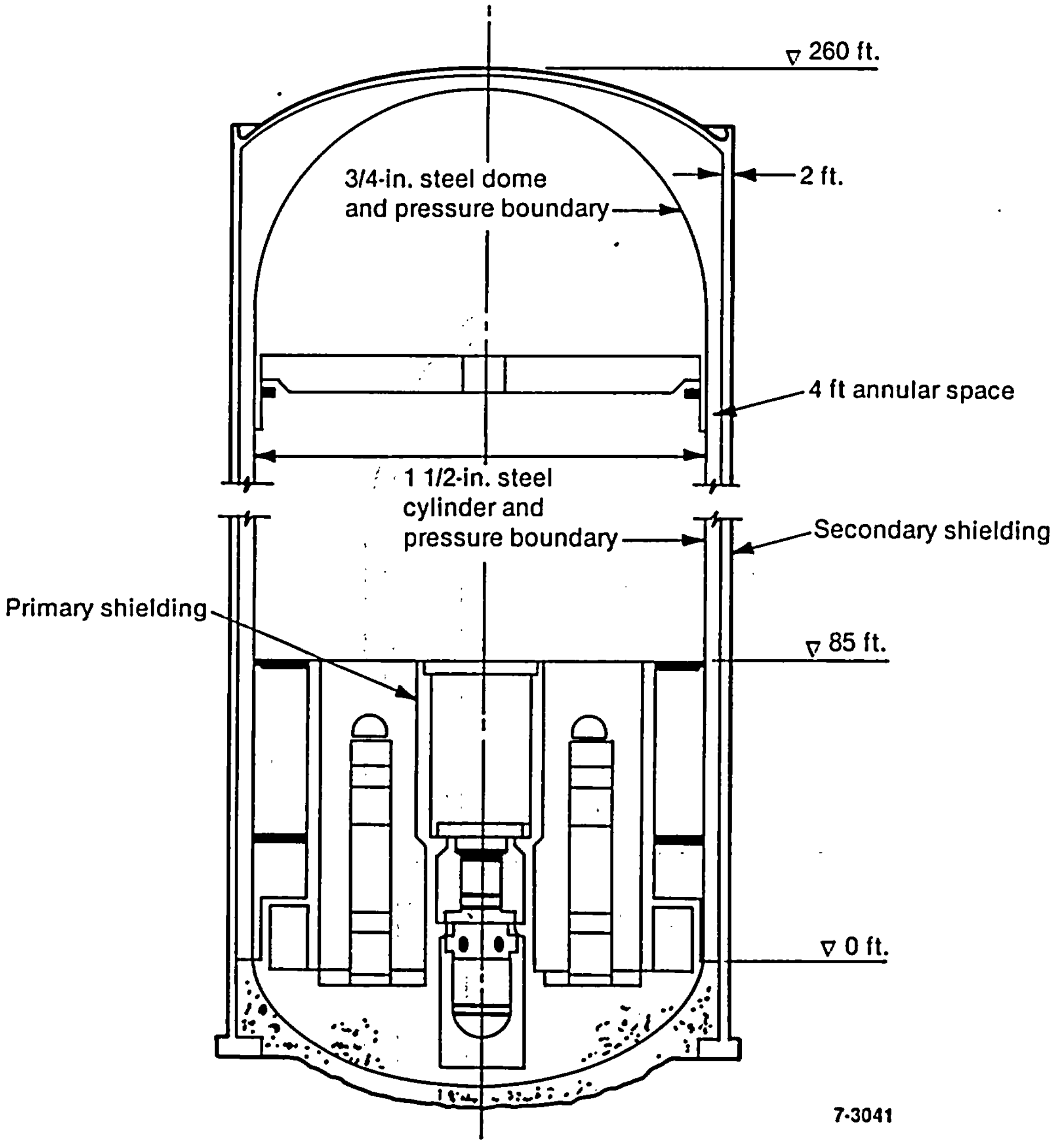

Figure 4.3. PWR steel cylinder containment with reinforced concrete reactor building.

are placed inside ducts within the containment wall and dome, which were formed during construction. The tendons provide prestressing forces in two orthogonal directions within the curved shell structure. For types I and II prestressed containments, [both types are shown in Figure 4.1(a)], the tendon anchorages are provided at the top and bottom of the cylinder for the vertical tendons, at six (Type I) or three (Type II) equally spaced buttresses on the circumference of the cylinder for the hoop tendons and around a ring girder at the top of the cylinder for the dome tendons. In Type III containments, [Figure 4.1(b)], the tendon anchorages are provided at the bottom of the cylinder and the vertical tendons extend from one side of the cylinder over the dome and down the opposite side. Tendon anchorages also are provided at three equally spaced buttresses on the circumference of the cylinder and extending vertically into the dome for the hoop tendon in the cylinder and the dome. All tendons and anchors are encased in a corrosion protective material and tightly capped at the ends.

The PWR containment structures are designed in accordance with the American Society of Mechanical Engineers (ASME) Section III, Division 2 code or equivalent requirements approved by the USNRC.[1] Material used in the construction of the containment structure is controlled by specific requirements specified by the American Society for Testing and Materials (ASTM) and other industry standards to guarantee a quality product. Quality control and quality assurance programs are set to control all steps of design, production, and construction of the containment.

The typical values of design/capability pressures and free volumes for the three PWR containments are as follows:

Containment (plant)	Design/Capability Pressure (psig)	Free Volume (10 E+03 cubic feet)
PCCV (Zion)	47/134	2,600
RCCV (Surry)	45/119	1,800
Steel cylinder (Sequoyah)	12/50	1,200

4.2 Containment Environment and Degrading Factors

Containment structures are subjected to a number of stressors that influence the characteristics and physical properties of the structural and nonstructural components of the containment. In general, exposure can be categorized as external, caused by the outside environment, and internal, caused by the environment created by the operating systems inside the containment building. An additional deteriorating factor unique to concrete is the adverse chemical reactions in concrete that are related to certain characteristics of the concrete mixture and its constituents.

4.2.1 External Environment. Exposure to the external environment affects mainly the concrete and in advanced cases of cracking and spalling it can possibly affect the embedded reinforcement. The adverse effects are due to wetting and drying, and freezing and thawing cycles. Acid rain, carbonation, salt spray on containment structures located near the sea, and harmful sulfates in ground water affecting the concrete foundation are the major degradation factors for both prestressed and reinforced concrete containments, and for concrete reactor buildings and steel shells in the case of steel containments.

The degree of deterioration caused by each of these factors varies, depending on the intensity of each and on the quality of the concrete and the corrosion protection system for the steel.

The external environment also affects the posttensioning system anchorage components and tendons mainly in the form of corrosion or hydrogen embrittlement caused by either rain or ground water seeping in, through the concrete or end anchorage caps, to the tendon ducts. The different degradation mechanisms are in most cases related to moisture conditions and accumulation of free water near the anchors.

These effects and degradation mechanisms will be discussed in detail in Section 4.4 of this report.

4.2.2 Internal Environment. The effects of the internal environment can be categorized as mechanical deterioration because of vibration loads, high temperatures, and fatigue stress and deterioration resulting from exposure to moisture and nuclear heating. The internal environment stressors affect both steel and concrete components in different degrees, depending on the severity of the stressor and the proximity of the components to the source of deteriorating factor. The posttensioning system is only slightly affected by the internal environment because of its location, embedded in the concrete wall. However, the steel liner may suffer from corrosion caused by the hot, moist atmosphere inside the containment, coupled with an acidic environment.

4.2.3 Chemical Reactions in Concrete. Aside from the external and internal environmental effects, concrete can be subject to adverse internal chemical reactions in the form of alkali-aggregate reactions, carbonate-aggregate reactions or cement-aggregate reactions resulting from adverse characteristics of the concrete mix ingredients.[2] Outside factors such as moisture and temperature conditions can enhance such reactivities. Reactivity in concrete may not occur or be noticeable for a long time after concrete placement and could vary in the degree of seriousness from insignificant scaling to complete spalling and deterioration.

4.2.4 Degradation Environment During Containment Service. Degradation factors also can be categorized as continuously acting stressors during normal operation of the plant, stressors occurring for short durations during accidents or malfunction of certain components, and as stressors naturally acting during the life of the plant.

Stressors during normal operation consist of radiation and nuclear heating, causing strength degradation and changes in physical and chemical properties of steel, concrete, and other materials such as gaskets and wires. Thermal stressors caused by elevated temperature, vibrating loads causing fatigue stresses, moisture and condensation causing corrosion, and acidic environments causing concrete and metal deterioration also exist during normal operation. In addition, normal operating procedures and scheduled pressure testing and refueling shutdowns can result in load and temperature cycling leading to fatigue and ratcheting effects. Naturally occurring stressors such as freeze and thaw, wetting and drying, acid rain, etc., may be considered as stressors occurring during normal operation. The degree of deterioration that may occur depends on how well concrete and other materials are constructed. The stressors that occur during accidents or malfunction of certain equipment can cause similar effects to those present during normal operation but at an accelerated rate and high magnitude that can possibly result in overstress conditions.

Structures are designed to a specified service life. Under normal conditions, most structures meet such a design criteria. However, beyond the design life span, degradation may alter the function, characteristics, and physical properties of components of the structure. Thus, reevaluation and possibly reanalysis of the load-bearing capability of the structural members becomes necessary using the properties of the members at the time of the reanalysis. Main stressors affecting the structures beyond the design life are those causing concrete deterioration, metal fatigue, and corrosion.

4.3 Degradation Sites

The degradation sites are best identified by examining each PWR containment type separately. However, some of these sites may be common to more than one containment type. The degradation sites for each containment type are identified in order of importance to the functionality and safety of each structure.

4.3.1 Prestressed Concrete Containment Vessels. The major structural components for the PCCV are the posttensioning system for the wall and dome, and the reinforced concrete wall, dome, and base slab as discussed above. The steel liner plate covering the internal surfaces of the containment, although it is not a structural load-bearing element, is important in providing a leak-tight barrier that will prevent radioactive releases.

Because the wall and dome are prestressed and prestressing induces a state of compression in both the steel liner and concrete, stress cracks in the steel liner are less likely to occur in this type of containment, and the possibility for release of radioactive material through the shell under normal operation is highly unlikely. Corrosion may develop in the liner or liner welds resulting in through-wall cracks. However, as long as the concrete is still in compression, the danger of radioactive material release is small. Therefore, degradation in the shell steel liner cannot be considered to be as important as loss of stress in the prestressing tendons. On the other hand, corrosion or cracks in the steel liner over the base slab can be important because radioactive water or other liquids may seep through the base slab and contaminate the ground water in the vicinity of the containment structure. Degradation in the base slab liner could become more significant if cracks were to have developed in the slab as a result of a seismic event or any other reason.

Loss of the prestressing stress (below the limits accounted for in the design) can be caused by broken wires or strands, cracked anchor heads resulting in partial or complete release of the tendons, excessive wire or strand relaxation or creep, and/or shrinkage in the concrete. Although the possibility of anchorage failure is remote, it has happened at various plants, after varying lengths of service times and different environments. This type of degradation can have important consequences because accident pressures inside a containment may cause the concrete shell and steel liner to enter a state of tension, at which time any cracks in the steel liner may possibly permit leakage of radioactive gases to the outside environment.[3] Therefore, the following degradation sites for prestressed containment are identified and ranked in the following order of importance:

1. Loss of prestressing force caused by:
   a. Loss of tendon anchorage
   b. Loss of tendon wires or strands
   c. Excessive tendon wire or strand relaxation
   d. Excessive concrete creep and shrinkage
2. Corrosion of the reinforcing steel embedded in the concrete

3. Cracks or corrosion in the steel liner over the base slab
4. Deterioration of the tendon corrosion protection material
5. Cracks or corrosion in the steel liner over the walls and dome
6. Deterioration in the concrete caused by the outside environment or internal reactions.

4.3.2 Reinforced Concrete Containment Vessel. RCCVs rely on reinforcing steel in the concrete to resist tensile stresses. With the absence of a prestressing system, the state of stress in the steel liner and concrete is not always in compression. Tension zones exist both in the wall, dome, and in the base slab. Therefore, cracks in the steel liner can be significant. Although seepage of radioactive material through cracks in the base slab is of first importance, leakage of radioactive gases through cracks in the steel liner on the wall and dome are also important. Because the tensile loads are carried by the reinforcing steel, corrosion of the reinforcement steel is of concern. Cracks and high porosity in the concrete cover over the reinforcing steel may permit moisture and oxygen to permeate and enhance the corrosion process. Therefore, good quality concrete is essential for both protection and strength. The possible degradation sites for the reinforced concrete containment can, therefore, be ranked as follows:

1. Corrosion of the reinforcing steel in the wall and dome
2. Corrosion of the reinforcing steel in the slab
3. Cracking and deterioration in the wall and dome concrete caused by the outside environment and internal chemical reactions
4. Cracks or corrosion in the steel liner over the base slab
5. Cracks or corrosion in the steel liner over the walls and dome
6. Deterioration in the base slab concrete caused by ground water sulfate and internal chemical reactions.

4.3.3 Steel Cylinder/Steel Sphere Containment Vessel. For this type of containment, the prime protection against leakage of radioactive gases is achieved by the same steel shell that also provides the structural barrier against pressure buildup inside the containment. Therefore, the most important degradation sites are associated with the shell itself. Degradation can be in the form of cracks or corrosion, either in the base metal or weld seams of the steel shell. Because the bottom part of the containment is embedded in concrete and the rest of the shell is freestanding, there exists a region of discontinuity at the top of the concrete where stress concentrations and vibrating loads in the steel shell may cause fatigue cracking. In addition, the same region is also susceptible to either stress-enhanced or galvanic corrosion. Other stress concentration zones exist in this type of containment in all other areas of discontinuities such as hatches and penetrations. Corrosion of the bottom portion of the containment that is embedded in the concrete is likely to occur and might eventually lead to escape of radioactive fluids to the ground outside the boundaries of the containment. The corrosion of this embedded portion could possibly be the result of attack by sulfates in the ground water, which reach the steel by penetrating through the outside concrete.

Elevated temperatures affecting the steel shell, either in areas near high-temperature equipment or at penetrations for high-temperature mechanical piping, can cause deterioration in the strength and other physical properties of the shell material.

The degradation sites for the steel cylinder/steel sphere containments can be ranked as follows:

1. Cracks or corrosion in the welds or base metal of the shell because of vibrating equipment and aggressive environments
2. Stress cracks or corrosion in the shell at the concrete embedment interface at the base of the containment
3. Stress cracks or corrosion in the shell at discontinuities near hatches and penetrations
4. Corrosion of the bottom steel of the shell embedded in concrete
5. Deterioration and cracks in the base slab concrete inside the containment
6. Regions of elevated temperature on the steel shell.

4.4 Degradation Mechanisms

The discussion of the potential degradation mechanisms is separated into three parts. Section 4.4.1 deals exclusively with the effects of radiation and nuclear heating on the physical properties of concrete. This is the only phenomena unique to structures that house nuclear reactors and associated systems. Several other phenomena that affect all concrete structures and are not unique to nuclear power plants are described in Section 4.4.2. For this reason, there is an abundance of research and literature on degradation caused by temperature, freezing-thawing cycles, sulfate attack, corrosion of the reinforcing material, etc. The potential degradation mechanisms associated with the various steel components are discussed in Section 4.4.3.

4.4.1 Effects of Radiation on Concrete. Because of the dual role of nuclear power plant concrete as a structural support and radiation shielding material, an evaluation of extended service life must address the continued effectiveness of the concrete in both its functions. From the standpoint of shielding, water is the most desirable ingredient in concrete, because water's high hydrogen content makes it an effective neutron moderator and attenuator. Because concrete generally serves as a structural support as well as a shield, the initial water content is maintained at a minimum because the structural strength of the concrete decreases with increasing initial water content. Thus, in the initial selection of the concrete constituents, shielding design considerations must be balanced with structural considerations. Similarly, the effects of aging and the various concrete degradation mechanisms must be evaluated in light of both shielding and structural requirements.

The most significant radiation degradation aspect of concrete results from nuclear heating. Nuclear heating produces an increase in concrete temperature causing evaporation of the free water in the concrete,[5] that is, water that is not chemically bound to the hydrated cement. Both the structural and shielding properties of concrete are somewhat affected by increased temperatures and loss of the free water.

As concrete temperatures increase, decreases in compressive strength, tensile strength, bonding strength, and the modulus of elasticity have been observed.[6,7] For gamma-ray shielding, the water loss as a result of nuclear heating has little impact because of the relatively small change in density associated with the loss. Because of the role hydrogen plays as a neutron moderator, the water lost at high temperatures can alter the neutron attenuation properties of concrete. However, the amount of the water loss during a 40-year operating life is not expected to be substantial enough to impair structural or shielding properties during extended operation.[7]

In contrast to the potential degradation of concrete resulting from nuclear heating and water loss, direct radiation damage is a less serious concern. This is because the fluxes that can cause direct radiation damage to concrete are higher by orders of magnitude than those that cause damage from nuclear heating. Hungerford[7] suggests that even a fast neutron exposure of 10^{21} nvt (n/cm^2) has little apparent effect on concrete aside from reducing its rupture strength by a factor of 2. Figure 4.4[8] shows the axial distribution of the fast neutron flux at the boundary of the reactor pressure vessel for a typical PWR. Because of attenuation, only the concrete closest to the surface directly exposed to the source is expected to approach a fast neutron exposure of 10^{18} nvt, which is well below expected damage levels.

Evidence of radiation-induced dissociation of the water (in small concrete samples) into hydrogen and oxygen, accompanied by gas evolution has also been reported.[7] This phenomenon is not expected to deplete the concrete's water content significantly, and, is most likely to occur near (i.e., within $\sim$1-1/2 to 2 feet of) the surface adjacent to the source.

Concrete samples obtained from the Nuclear Waste Tank Farm built in 1953 (Rockwell Hanford Operations, Richland, Washington) have been tested for the effects of long-term exposure to temperature and radiation.[9,10,11] The results indicated limited reduction in strength and modulus of elasticity but did not indicate any reduction in the integrity of the concrete. Thus, even in extended plant operation, degradation of concrete because of radiation is not expected to be life limiting.

4.4.2 Effects of Other Concrete Degradation Mechanisms. The degradation of reinforced concrete can be described in terms of elements affecting the deterioration of the concrete and the subsequent corrosion of the reinforcing steel after its protective cover is degraded. The potential degradation of concrete can be divided into two categories. The first category of potential degradation is related to the initial material selection, proportioning and mixing procedures, and placement and curing of the concrete. The second category of potential degradation is related to the environment to which the concrete will be subjected during the life of the plant.

4.4.2.1 Initial Potential Degradation of Concrete. This category of potential degradation is influenced by such factors as physical properties of

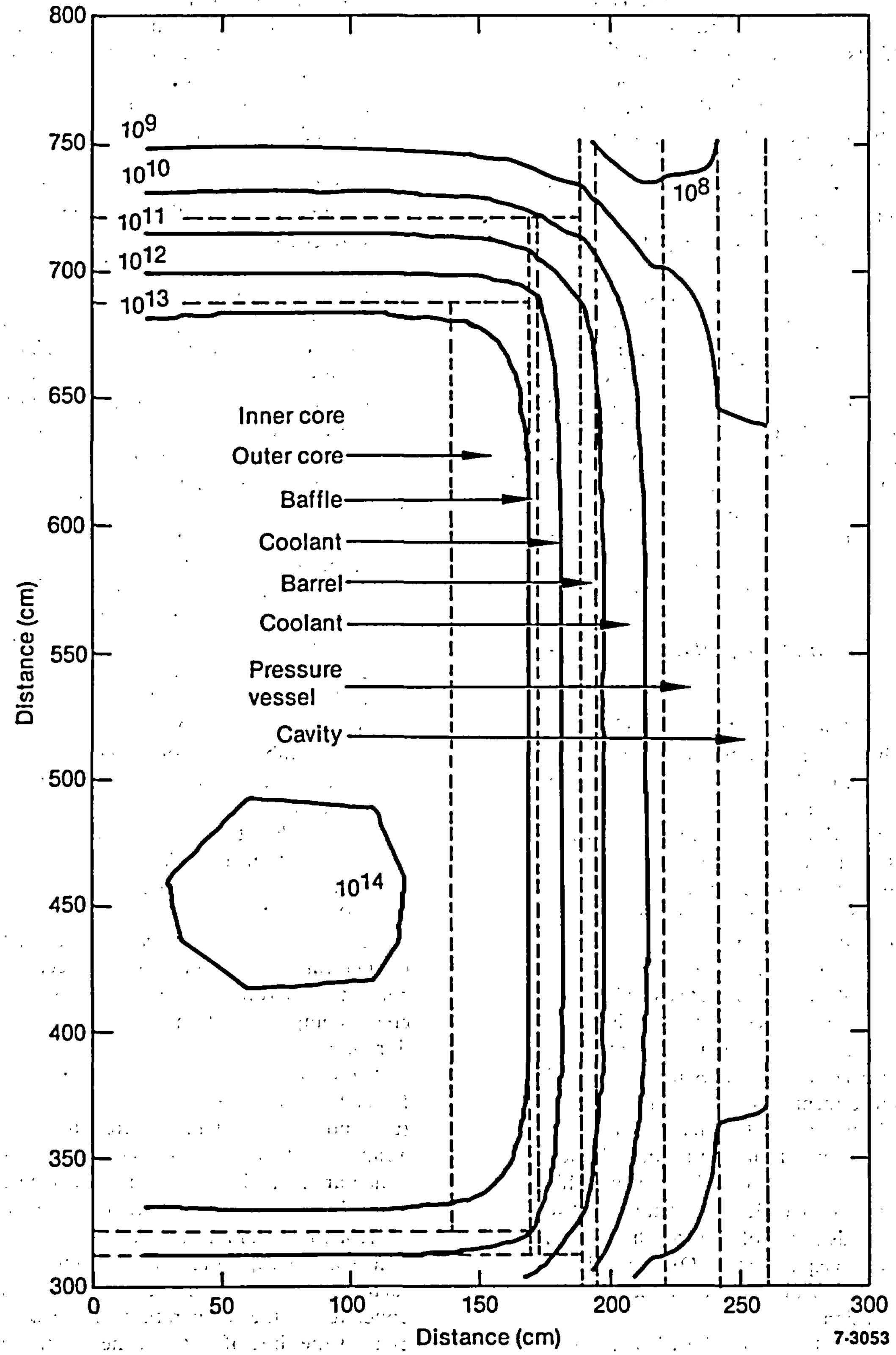

Figure 4.4. Isoflux contours for neutrons with energy > 1 MeV in the PWR (R,Z) model (neutrons/cm^2 s).[9]

the aggregate, water-cement ratio, type and quantities of cement, shrinkage and settlement cracking, temperature effects related to heat of hydration, and adverse chemical reactions. These potential degradations can be controlled by the proper choice of materials, mix ratios, and curing procedures. However, if any of the key parameters affecting the concrete integrity are not handled properly during mixing and placing, the quality of the concrete can be reduced which will increase the potential for later degradation.

These degradation modes usually result first in surface cracking of the concrete. Cracking of concrete can occur while the concrete is in either a plastic or hardened state. The presence of cracks normally does not affect the ability of the structure to carry load, but can affect structural durability by exposing the embedded reinforcements to a corrosive environment. Excessive shrinkage resulting in cracks of unacceptable sizes usually results from use of materials and techniques that are not in accordance with good engineering practice and that would not pass standard construction inspections.

4.4.2.2 Environmental Degradation of Concrete.

1. *Chemical Reactions.* A number of deleterious chemical reactions can produce concrete cracking. These generally are related to the concrete aggregate and include: alkali-aggregate reactions, cement-aggregate reactions, and the reaction of carbonate aggregates. Sulfate-bearing waters also present problems, but by attack of the cement paste rather than involvement of the aggregate.
   a. *Alkali-Aggregate Reactions.* Cracking can occur as a result of expansive reactions between aggregates containing active silica and alkalies usually derived from cement. Problems with structures built since about 1950 have been significantly reduced through proper selection of aggregate materials (petrographic examination to identify potentially active materials), use of low alkali cements (<0.6% equivalent Na_2O), and addition of pozzuolanic materials that restrict expansion.
   b. *Cement-Aggregate Reactions.* Concrete deterioration attributable to cement-aggregate reactions has been observed with some sand-gravel aggregates. These materials are highly siliceous and produce map cracking in concrete. This type of distress is not prevented by the use of pozzuolana or low-alkali cements. Tests have been devised to indicate potential damage from this phenomenon.[2] A recommended remedy, if these materials must be used, is to dilute the aggregate with crushed limestone.
   c. *Carbonate-Aggregate Reactions.* Certain dolomitic limestone aggregates containing some clay react with alkalis to produce an expansive reaction. The alkali-carbonate reaction can be controlled by keeping the alkali content of the cement low (<0.4%) or by diluting the reactive aggregate with a less-susceptible material.
   d. *Sulfate-Bearing Waters.* The sulfates of sodium, potassium, and magnesium have caused deterioration of many concrete structures though chemical reaction of the sulfates with the hydrated lime and hydrated calcium aluminate in the cement paste. Calcium sulfate and calcium sulfoaluminate are formed along with considerable expansion that disrupts the concrete. Resistance of concrete to sulfate attack can be improved by the use of admixtures such as pozzuolana and blast furnace slag or the use of sulfate-resisting cements (Types V, VP, VS, IP, or IS).[12]
2. *Elevated-Temperature Effects.* Elevated temperature has an important effect on concrete structures because it affects concrete's compressive strength, stiffness, and durability.

   A review of research and testing performed on concrete at elevated temperatures indicates that temperature effects are a function of a number of factors, some of which are related to the type of aggregate and mix proportions and others of which are related to the limits and duration of the temperature exposure. Other important factors influencing the results of tested concrete are the temperature of the concrete at the time of test and the condition of the concrete during heating (whether sealed or open). Thermal cycling is also important because it can cause cracking and separation between the concrete matrix and aggregate and between the concrete and the reinforcing steel.

The effects of the type of aggregate on compressive strength do not become significant until the temperature reaches ∿600°F (316°C). Neville[2] reported that the compressive strength exhibits a slight increase between 400 and 600°F (204 and 316°C) (Figure 4.5). The modulus of elasticity is reported by Neville to drop with increased temperature, Figure 4.6.

Bertero and Polivka[13] concluded that moisture in sealed specimens has a detrimental effect on concrete exposed to high temperature. They reported a 71% and 51% reduction in compressive strength and elastic modulus, respectively in sealed specimens exposed to 300°F (149°C). However, other specimens from the same concrete experienced insignificant reductions in mechanical properties when moisture was allowed to escape during the thermal exposure.

3. *Wetting and Drying.* In structures with cracks, improperly treated construction joints, or areas of segregated or porous concrete, water may enter and exit. As water passes through the concrete, it dissolves (leaches) some of the calcium hydroxide and other solids so that in time serious disintegration of the concrete may occur. As the concrete dries, salts (sulfates and carbonates of sodium, potassium, or calcium) are deposited through evaporation of the water or interaction with carbon dioxide in the atmosphere. Extensive leaching causes an increase in concrete porosity, which leads to reduced strength and increased vulnerability to other corrosive environments.
4. *Freezing and Thawing.* Repeated cycling of wet concrete between freezing and thawing can lead to severe damage or destruction of concrete structures. The principal force responsible for concrete damage resulting from this phenomenon is internal hydraulic pressure created by the expanding ice-water system during freezing. Construction of highly freeze-thaw resistant structures requires attention to these factors: (a) use of air entrainment, (b) selection of aggregate with durability adequate for the proposed exposure, and (c) use of initial low water-to-cement ratio concrete.
5. *Cracking.* Cracks in concrete surfaces can be indicative of normal processes such as drying shrinkage or of unusual (and unwanted) processes such as foundation settlement, structural overload, chemical reactions, and other degrading processes. Because the opening of cracks allows access by hostile environments to reinforcing steel and other embedded items, it is important to investigate cracks of significant size.
6. *Aggressive Chemical Environment.* Aggressive environments of high temperature, moisture, and acid-saturated fluids or gases are common in power plants. Such environments are detrimental to both concrete and steel containment types. When concrete is exposed to an acidic environment, a reaction occurs between the acids and the calcium hydroxide of the hydrated portland cement. Some of the acid reactions result in water-soluble calcium compounds that are then leached away by aqueous solutions. In the case of sulfuric acid, the deterioration is more pronounced because the reaction product occupies a larger volume than the acid and the calcium hydroxide, resulting in spalling and cracking in the concrete.
7. *Corrosion of Embedments.* Cracking and spalling of concrete can result from the corrosion of embedments, primarily mild steel reinforcing bars. Concrete cracking and spalling results from the buildup of corrosion products that can act as a wedge to delaminate and spall the adjacent concrete.

   Continuous exposure of aluminum to moist concrete produces severe corrosion because of the destruction of its passive film at high alkalinities. Lead and zinc also corrode in moist concrete, with zinc being more resistant than aluminum or lead. In all these cases, the corrosion process is aggravated in the presence of chloride ions.
8. *Vibration and Structural Loads.* Concrete and steel can both fail in fatigue. After a few hundred thousand reversals at a significant stress level, the concrete strength can be reduced to half its original value. The effects of fatigue on concrete are more pronounced if the stress reversals alternately place the concrete in tension and compression.[2,4] The

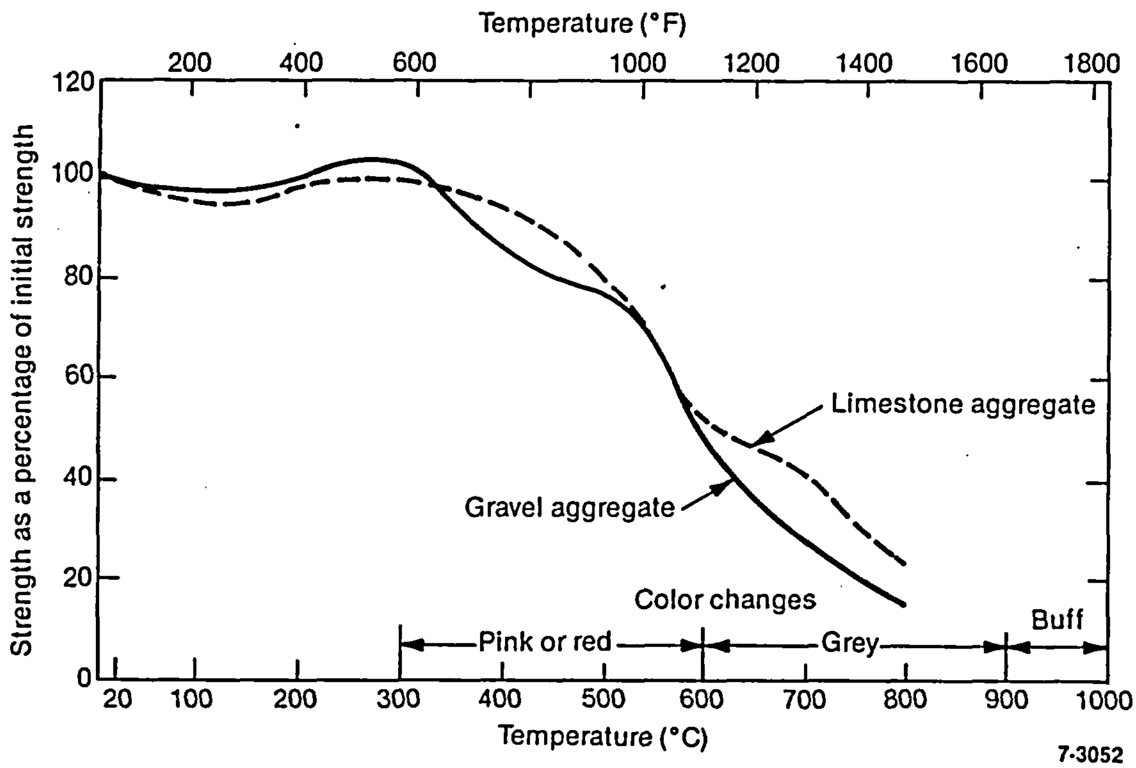

Figure 4.5. Compressive strength of concrete after heating to high temperatures.[3]

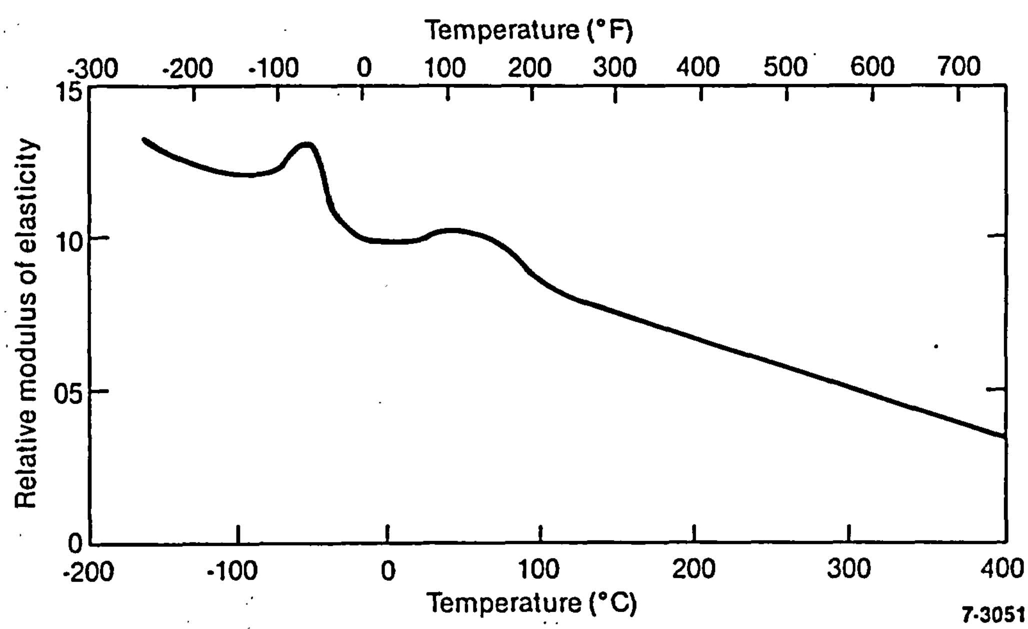

Figure 4.6. Influence of temperature on modulus of elasticity of concrete.[3]

durability of the concrete can be improved significantly if the stress reversals are such that the concrete always remains in compression.

Excessive vibration can lead to cracking of the concrete. The presence of such cracks may expose the reinforcing steel to corrosive elements.

4.4.3 Steel Degradation Mechanisms and Their Effects. This section will discuss the potential degradation mechanisms associated with reinforcing steel, posttensioning system steel, steel liner plate, and steel material used for the steel cylinder/steel sphere containment shells. Unlike concrete, the major degradation mechanism for steel is corrosion. Elevated temperatures and radiation also have the potential to degrade steel.

4.4.3.1 Corrosion of Steel. The potential for corrosion of the reinforcing steel is highly dependent on the quality of the concrete in which it is embedded and on the presence of cracks and porosity in the concrete that may allow moisture and other harmful substances to reach the steel. Corrosion, when it occurs, will result in iron oxides that expand in volume causing spalling of the concrete and further exposure, allowing more corrosion to occur. When a steel containment is exposed to an acidic environment and protective measures such as coatings or a dry environment are not provided (or have worn out), the acid attacks the steel. This results in chemical corrosion. Scaling of the corrosion product will result in a reduced structural section leading to increased stress intensities and reduced leak-tightness capability. Similarly, corrosion of the steel liner in concrete containments could jeopardize its leak-tightness capability, which is the primary function of the liner.

Corrosion in posttensioning system components may be either localized or uniform. Most prestressing, corrosion-related failures have been the result of localized corrosion in the form of pitting, stress corrosion, or hydrogen embrittlement. A review of the prestressing tendon surveillance reports indicates that corrosion incidents are extremely rare. However, corrosion has been detected at a number of plants during the tendon system inspection. The most common incidents of corrosion are pitting on the tendon wires or strands. The pitting in most cases has been limited and attributed to the presence of small amounts of water that have accumulated at the lower end of the vertical tendons in the area of the anchorage. This problem has been encountered in Types I and II posttensioned PWR containments, Figure 4.1(a), where the configuration of the vertical tendon anchorage possibly allows some rain water ponding around the dome to seep through the top and travel to the bottom anchorage. Another reason for pitting corrosion of the wires and anchor heads may have been a poor construction practice where the tendons were either stored at the site without proper protection for long periods or were not properly protected after installation and before application of the permanent corrosion protection material.

A few anchor heads on different containments have experienced either cracking or partial failure and in one case a complete failure because of hydrogen embrittlement. The time at which these failures occurred ranged from a few days to a few years after stressing. The chemical composition of the anchorhead material and heat treatment procedures used are thought to have contributed to the failures. It has not been positively confirmed whether such failures are time dependent and whether they are related to a specific environment near the anchorage.

4.4.3.2 Effects of Elevated Temperatures on Steel. In general, elevated temperature decreases the yield strength and modulus of elasticity of the steel and increases its ductility. However, these effects are only pronounced when the steel is exposed to temperatures of ∼300°F (149°C) and above.[14,15] One characteristic of special importance, even at temperatures of 100 to 150°F (38 to 66°C), is the relaxation of the posttensioning tendon steel that increases with increased temperature resulting in loss of the prestressing force. Fortunately, the tendons are located in the containment shell within the concrete which acts as an insulator and protects the tendons from exposure to harmful temperatures.

4.4.3.3 Effect of Radiation on Steel. This is not a potentially damaging mechanism for reinforcing bars, liner, structural steel, and posttensioning tendons. Reinforcing steel and posttensioning tendons in the containment shell are normally exposed to limited amounts of radiation compared to the structural steel inside the containment. Exposing test specimens to radiation flux levels higher than that which posttensioning tendons are normally exposed to has resulted in harmless effects.[7] Therefore, this mechanism is not important.

4.5 Potential Failure Modes

Possible containment failure modes during normal operation, accidents, and postdesign life are discussed in this section.

4.5.1 Prestressed Concrete Containment Vessels

4.5.1.1 Failure Modes During Operation. Structural failure under load is not anticipated during normal operation because load-carrying members are designed with considerable margin to carry such applied loads. Therefore, any failure modes are expected to be the result of exposure to adverse environmental conditions. The failure modes are ranked in order of importance as follows:

1. Loss of prestressing force. There are a number of factors that can lead to different levels of losses in the prestressing force. The most severe occurrence is the failure of an anchor, resulting in either partial or complete loss of the tendon stress. In addition, dealing with the situation of a partially failed anchor presents a safety hazard to workers replacing the damaged anchor. In most cases, failure of anchors has been caused by hydrogen embrittlement resulting from either improper chemistry or improper heat treatment of the anchor materials. Failure of individual wires or strands in the tendons is also of concern. This type of failure can be caused by either overstress in individual wires or strands, localized inclusions of weak material, localized brittle fracture because of inclusion of an embrittling type of material, or reduction in cross section because of pitting and corrosion. Corrosion in posttensioning components has occasionally occurred and may be the result of breakdown in the effectiveness of the corrosion protection material (grease) around the tendons and anchor components or the presence of moisture and water in areas around the tendons and anchors. Microbiologically influenced corrosion of the carbon steel tendon wires has been reported at some PWR sites.[16] Prestressing losses caused by creep and shrinkage of the concrete and relaxation of the tendon material might also result in failure of the posttensioning system, if the losses exceed those accounted for in the design.
2. Corrosion of the reinforcing steel. This mode normally occurs as a result of poor concrete quality or concrete deterioration. The consequences can be serious and repairs are complicated.
3. Corrosion of the steel liner over the base slab. This mode of failure can lead to leakage of radioactive fluids through the liner into possible cracks or porosity in the concrete slab and out to the outside environment.
4. Deterioration of the concrete. During the life of the plant, the concrete may be subjected to the degradation mechanisms described in Sections 4.4.1 and 4.4.2. The resistance of the concrete to degradation is related to the quality of the concrete and the construction procedures used. Deterioration of concrete can have a number of side effects, ranging from serious structural consequences to superficial scaling. The primary shield wall is among the concrete structures that should be inspected.

4.5.1.2 Failure Modes During Accidents. Accidents which could adversely affect containment structures are those which result in large increases in pressure or a hydrogen burn explosion inside the containment. In the case of prestressed concrete containments, the hoop stresses in the vessel control the failure modes. Resistance of the containment to this type of loading is dependent on the adequacy of the prestressing system. When the internal pressure generated by the accident exceeds twice the design pressure, it could cause sizable cracks in the concrete while structural stability is still maintained. A liner-concrete interaction may take place at locations of major concrete cracks and distortions and introduce high strain concentrations that may lead to localized rupture of the steel liner plate provided the liner is firmly bonded to the concrete. The ruptured liner plate will then allow some leakage of radioactive gases to the outside environment. This failure mechanism has been proposed by Rashid and is sometimes referred to as the Rashid mode of failure.[17] Other failure modes have been identified for containment failure at pressure levels higher than design pressure.[18]

4.5.1.3 Failure Modes Beyond Design Life of Containment. Most of the possible failure modes for this stage are similar to the failure modes possible during normal operation. However, problems associated with the loss of the prestressing force because of creep and shrinkage of the concrete and relaxation of the tendon material may become more important because these changes, with time, will eventually consume all or most of the design margin provided. Load cycling resulting from scheduled and forced shutdowns, and integrated leak rate tests, may also develop fatigue cracks that jeopardize the integrity of the containment components.

4.5.2 Reinforced Concrete Containment Vessels. The potential failure modes for RCCVs are similar to those for prestressed concrete containments except for the failure modes related to the posttensioning system (that does not exist in the reinforced concrete containment). However, reinforced concrete containment failure in the hoop direction under accident pressures can be more pronounced than in the prestressed concrete containment. RCCVs develop widespread cracking at about 75% of the design pressure because of the absence of the prestressing forces. Therefore, localized strains in the steel liner plate due to liner-concrete interaction can be large and may lead to leakage of radioactive gases.

4.5.3 Steel Cylinder/Steel Sphere Containment Vessels. The loads in the steel containment structures are entirely resisted by the structural steel of the shell. Cracks and corrosion in welds and base metal of the shell dominate the possible failure modes in this type of containment structure during normal operation. Containment cracks present both structural and safety hazards because of stress concentrations and leakage of radioactive gases, respectively. Corrosion and fatigue are the potential failure modes for operation beyond the design life. Of special concern are the shell areas near discontinuities such as the base of the shell, hatches, and penetrations.

4.6 In-service Inspection

In-service inspection (ISI) of the critical components of the containment structures are a mandatory part of the operating licenses of nuclear power plants. Each containment is subjected to a structural integrity test (SIT) and an integrated leak rate test (ILRT) before operation of the plant to ensure complete leak tightness and pressure resistance capability. The SIT and ILRT consist of pressurizing the containment to 15% above its design pressure. Associated with the SIT are measurements of containment growth under pressure for evaluation of actual containment behavior in comparison with design predictions. The ILRT includes instrumented and visual examination of surface-crack initiation and growth at critical locations such as hatches, the ring girder, and the posttensioning system anchorage regions. After the ILRT, the surface of the steel liner plate also is visually inspected for any cracks, separation, hollow sounding or inward buckling regions, and corrosion that might indicate any degradation in the system. A brief description of the visual examination is given in Chapter 11. The ILRT is repeated every five years to ensure structural integrity and continued safety of the containment. Additional discussion on the visual examinations and leak testing of containments is given in Chapter 12.

In addition to the ILRT tests, ISI of the posttensioning system in the prestressed concrete containment vessels is performed one, three, and five years after the initial SIT-ILRT inspection and every five years thereafter. This inspection is required by the operating license and ASME Section XI and is conducted to monitor the level of prestressing forces in the posttensioning tendons and inspect the condition of the tendons, anchorages, shims, bearing plates, and corrosion protection materials. In addition, wire or strand samples are obtained during each ISI from one or more tendons and tested for tensile strength. The concrete surface condition near the anchorages, where it is stressed the most, is visually examined for the presence of any cracks or deterioration. The ISI surveillance is also expected to identify any adverse effects related to tendon relaxation or creep of the concrete.

Problems identified during these tests are normally corrected immediately. Gaskets, seals, welds, etc. are provided when measured leak rates exceed the standard design allowables. Prestressing levels are restored when tendon losses exceed those predicted. Tendons, anchors, and corrosion protection material are replaced when corrosion or cracking is observed on the tendons or anchors and when chemical and physical test results indicate any degradation in the corrosion protection materials.

In addition to the SIT, ILRT, and ISI, a walkdown and inspection of all components, supports, structural elements, and system trains is performed during each refueling outage. A punch list is generated and executed for correction of deficiencies discovered during the walkdown.

A great deal of information is collected on the condition of each containment as a result of all the above described inspection and testing programs. However, in light of the degradation mechanisms described in Section 4.4 and the failure modes postulated in Section 4.5, additional monitoring may be required to maintain the containment in a reliable condition beyond the design life.

General monitoring and ISI procedures for extended life operation should be established to identify the damage due to any likely and significant degradation mechanisms in a timely manner. Such procedures may include both nondestructive and destructive test methods. Nondestructive methods may consist of visual inspection, ultrasonic inspection and testing, radiography, infrared thermography, rebound method for surface hardness, etc.[19,20] Destructive methods should be used only when the nondestructive methods are not sufficient to quantify the extent of the degradation damage. Destructive methods may consist of obtaining core samples from the in situ concrete for compressive-strength tests and petrographic examinations,[21] or the use of penetration techniques such as the Windsor Probe and pullout testing for determination of in situ concrete strength. Core samples from the degraded regions would also provide a direct means to evaluate the width and depth of cracking or the extent of any voids. Steel samples from both reinforcing steel and structural steel components also may be required for laboratory testing.

Establishment of inspection procedures to cover critical areas where adverse environmental conditions such as high temperature, humidity, and/or radiation, and locations subjected to an acidic environment, will be a necessary measure to determine the extent of degradation. Representative samples of steel and concrete should be obtained from critical areas and tested to determine the existing chemical and physical properties of the materials. Specifically, inspection of concrete against deterioration from environmental factors and underground water attack and monitoring of concrete compressive strength by testing some core samples may be necessary. Specific examination of steel members for corrosion, cracking, and distress because of cyclic loading conditions may also be necessary.

In summary, by implementing a well planned, meaningful, and comprehensive inspection program followed by a firm program of repairs and maintenance, the service life of containment structures can be extended, probably to at least double the initial licensed term and possibly much longer.

4.7 Summary, Conclusions, and Recommendations

The nuclear power plant containment degradation sites include areas and components where degradation may cause either structural damage or a safety hazard or both. Degradation mechanisms unique to nuclear power plants such as those resulting from radiation and nuclear heating and elevated temperature have been discussed in detail. Other degradation mechanisms that affect general structural components have also been discussed.

The available research and test results indicate that radiation and nuclear heating have little effect on the shielding and structural capability of concrete. However, activation of primary shield concrete surfaces in the reactor vessel cavity by neutrons, even to a few inches in depth, can be a real radiation protection problem after 40 years of operation.[22] The effects of radiation on the steel used in PWR containments appear to be insignificant within the dose range experienced during normal operation of the plant. Although it appears that elevated temperature does affect concrete, its degradation effects are unique to each plant based on the concrete material and quality used.

Degradation caused by freezing and thawing, wetting and drying, and chemical reactions is a condition that is related to the quality and durability of the concrete selected for the plant. Careful testing, evaluation, and selection of the concrete material with proper mix proportioning and construction procedures could minimize or eliminate degradation related to these factors. However, some remedial measures may be taken to limit or slow the degradation process if it occurs. Such measures can be in the form of actions to prevent moisture and free water from entering and leaving the concrete.

Corrosion of the structural or reinforcing steel can be a major degradation mode; however, it should be noted that corrosion is initiated as the result of adverse environmental conditions. Distress caused by corrosion whether in structural steel or reinforcing steel can have a great structural impact. Cracks and fatigue caused by vibrating loads and dynamic loads can be serious in terms of the structural integrity of the affected members.

Failure modes are closely related to the described degradation mechanisms. The failure modes postulated are general because they are presented on a generic basis. More unique failure modes such as liner-concrete interaction are expected to be identified when evaluations are performed on a plant-by-plant basis.

The importance of ISI has been recognized and reflected in the presently imposed inspection programs for the containment structures and components, in particular, the posttensioning system for the prestressed concrete containment vessels. It should be realized that great safety and economic benefits can be derived if an expanded ISI is implemented to cover identified degradation sites that may not be frequently inspected.[23] Immediate repair or remedial action is of great importance in arresting any degradation before it progresses to a point where other components begin to degrade or other degradation mechanisms start that otherwise would have not started. A simple example of this is deterioration of the concrete cover over the reinforcement. If deterioration of the cover material is promptly detected and repaired, degradation of the reinforcing steel will not occur.

Tables 4.1, 4.2, and 4.3 provide a summary of the degradation sites, stressors, mechanisms, failure modes, and necessary inspection methods for the three types of the PWR containments discussed in this report.

There are several issues associated with aging and life extension of the PWR containments that need to be resolved. A comprehensive and standardized ISI program is needed to identify and quantify degradation in reinforced concrete. The state-of-the-art methods used in Europe, Japan, and the U.S.A. for evaluating the condition of reinforced concrete need to be assessed and included in the program. A standardized program implies the same ISI program for all LWRs. The ASME Section XI committees on metal and concrete containments are evaluating the current ISI methods for reinforced concrete and developing specific requirements for thorough visual inspections.

Additional information about the long-term degradation of reinforced and prestressed concrete containments is needed. Data should be collected from the older LWR containments. Also, facilities that have been shut down after an extended period of service (such as Shippingport, Dresden, or Humbolt Bay) could be used to obtain additional aging-related data on reinforced concrete materials. Accelerated aging techniques could also be investigated and, if found to be appropriate, used to obtain posttensioning system and concrete materials aging data. The results from these investigations will contribute to the development of an aging-related material property data base and support the development of quantitative models of the degradation of reinforced concrete subjected to long-term exposure to elevated temperatures, radiation, and cyclic loadings. Two items in the posttensioning system that need particular attention are tendon relaxation and degradation of the grease used as the corrosion inhibitor. Also, a monitoring method is needed to detect degradation of the anchorages because of hydrogen embrittlement and aging of the grease caused by impurities. Other less important recommendations include evaluation of the potential interactions between chemical reactions in concrete and radiation, and determining the extent of the deterioration of the shielding properties of concrete caused by radiation heating.

The structural integrity of aged metal containment and analysis of concrete-liner interactions in aged RCCVs and PCCVs during severe accidents are two other issues that need to be resolved. The results from the testing of containment models subjected to pressure and thermal loadings and the results from the EPRI-sponsored research programs to determine the significance of concrete-liner interactions in aged containments will be helpful.

Table 4.1. Summary of degradation processes for prestressed concrete containment vessel

Rank	Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes	ISI Method
1	Posttensioning system anchorage	Material properties and trapped water	Hydrogen embrittlement	Loss of stress	Tendon surveillance program
2	Posttensioning tendon wire or strand	Moisture, trapped water, or breakdown of grease material	Pitting, microbiological-induced corrosion	Loss of stress	Tendon surveillance program
3	Steel liner dome and wall	Moisture, acidic environment, and stress	Corrosion	Liner-concrete interaction, leakage of radioactive gases	Leakage testing (10 CFR 50, Appx. J)
4	Steel liner over base slab	Moisture, acidic environment, and stress	Corrosion	Leakage of radioactive material	Leakage testing (10 CFR 50, Appx. J)
5	Dome, wall, and base slab reinforcing steel	Aggressive environment	Corrosion	Loss of structural integrity	Visual
6	Concrete	Aggressive environment and internal chemical reactions	Cracking and spalling	Loss of integrity, corrosion of reinforcing steel	Visual, rebound methods, core samples if required

Table 4.2. Summary of degradation processes for reinforced concrete containment vessel

Rank	Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes	ISI Method
1	Dome and wall reinforcing steel	Aggressive environment	Corrosion	Loss of structural integrity	Visual
2	Base slab reinforcing steel	Aggressive environment	Corrosion	Loss of structural integrity	Visual
3	Steel liner over dome and wall	Moisture, acidic environment, and stress	Corrosion	Liner-concrete interaction, leakage of radioactive gases	Leaking testing (10 CFR 50, Appx. J)
4	Steel liner over base slab	Moisture, acidic environment, and stress	Corrosion	Leakage of radioactive material	Leakage testing (10 CFR 50, Appx. J)
5	Dome and wall concrete	Aggressive environment, internal chemical reaction	Cracks and spalling	Loss of integrity, corrosion of reinforcing steel	Visual, rebound methods, core samples if required
6	Base slab concrete	Aggressive environment, internal chemical reactions	Cracks and spalling	Loss of integrity, corrosion of reinforcing steel	Visual, rebound methods, core samples if required

Table 4.3. **Summary of degradation processes for steel cylinder/steel sphere containment vessel**

Rank	Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes	ISI Method
1	Shell welds and base metal	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Loss of structural integrity, leakage of radioactive gases	Visual, leakage testing
2	Interface between shell and concrete slab at base of shell	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Loss of structural integrity, leakage of radioactive gases	Visual, leakage testing
3	Discontinuities in the shell such as hatches and penetrations	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Leakage of radioactive gases	Leakage testing (10 CFR 50, Appx. J)
4	Steel bottom of shell embedded in concrete	Aggressive environment	Corrosion	Leakage of radioactive material	Leakage testing (10 CFR 50, Appx. J)
5	Base slab concrete	Aggressive environment, internal chemical reactions	Cracks and spalling	Corrosion of reinforcing steel, corrosion of steel bottom of containment shell	Visual, rebound methods, core samples if required

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5. PRESSURIZED WATER REACTOR COOLANT PIPING

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5.1 Description

5.1.1 Geometry, Design, Vendors. Reactor coolant piping systems for the three principal pressurized water reactor (PWR) types are shown in Figures 5.1, 5.2, and 5.3. The PWR designers are Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse (W).

As can be seen, the B&W and CE designs are similar in that there are two heat exchange loops. Each loop has a hot leg from the reactor to the steam generator, and two cold legs that return the coolant from the steam generator to the reactor. There are four pumps, one in each cold leg.

The only major conceptual difference between the CE and B&W design is that the B&W steam generator (SG) is a once-through straight tube type, and the CE design is a recirculating U-tube type. Circulation of primary coolant in the U-tube SG is from the bottom hot-leg side up to the top of the tube bundle, then down to the bottom of the cold-leg side. The B&W straight-tube SG path for the primary coolant flow is from the top head vertically downward to the bottom head.

Consistent with the SG designs, the primary system hot-leg piping for the B&W design goes from the reactor vessel outlet nozzle upward to the top of the SG. The hot-leg piping for the CE design goes directly across the compartment from the reactor vessel outlet nozzle to the bottom head of the SG. The overall arrangement of the cold-leg piping for these two designs is quite similar.

The third major PWR design, W, is conceptually different from the CE and B&W designs in that each heat exchange loop has a SG and a reactor coolant pump. A given reactor plant may have two, three, or four heat exchange loops. The number of loops in any given plant is dependent on a number of factors, but the most important is the capacity of the plant. Each loop increases the amount of power that may be extracted from the reactor.

5.1.2 Materials and Fabrication. The later W plants have main coolant piping made from centrifugally cast austenitic stainless steel, and the fittings are statically cast stainless steel. However, the main coolant piping in the early plants was wrought austenitic stainless steel (such as in Surry-1). The main coolant piping for the CE and B&W plants is wrought ferritic steel clad with stainless steel. The materials for all piping are shown in Table 5.1.

The ferritic piping used by B&W and CE is clad with austenitic stainless steel, as noted in Table 5.1. B&W uses weld-deposited cladding, and CE uses roll-bonded cladding. B&W uses explosively bonded cladding in the piping elbows. The elbows are fabricated with two half elbows (clam shells) and the clad is applied before the two halves are welded together.

There is no need for cladding in the W pipe because it is entirely stainless steel. As noted in Table 5.1, however, there are special circumstances with the weld between the austenitic pipe and the ferritic reactor vessel and steam generator. Because any weld to the ferritic steel would require stress relief in these wall thicknesses, a layer of Inconel 600 is deposited on the nozzle ends before final vessel stress relief. Then, the stainless steel pipe can be welded to the Inconel and no additional stress relief is required. The Inconel was not thought to become sensitized in the vessel stress relief.

CE piping is constructed of roll-bonded cladded plates. The plates that are cladded during the rolling process are formed into rounds and seam welded. Careful weld preparations permit austenitic weld material to be used on the cladding and ferritic weld material on the ferritic plate forming the main pipe wall.

5.2 Stressors

Nuclear plants are most economical when they operate at steady-state, full power from one refueling to the next. Certainly, all plant owners strive for this type of operation. From a technical point of view, this mode of operation is also optimum. The only significant source of stress is the internal pressure. In the steady-state condition, thermal stresses do exist because of minor degrees of restrained expansion, but they are minimal.

However, for one reason or another, nuclear plants rarely operate for a complete fuel cycle at full power in an undisturbed condition. The concern about safety is so great that the plants are frequently shut down and examined for off-normal situations. Equipment failures often require shutdown to cold conditions. For various reasons, reductions in power are common. Testing of

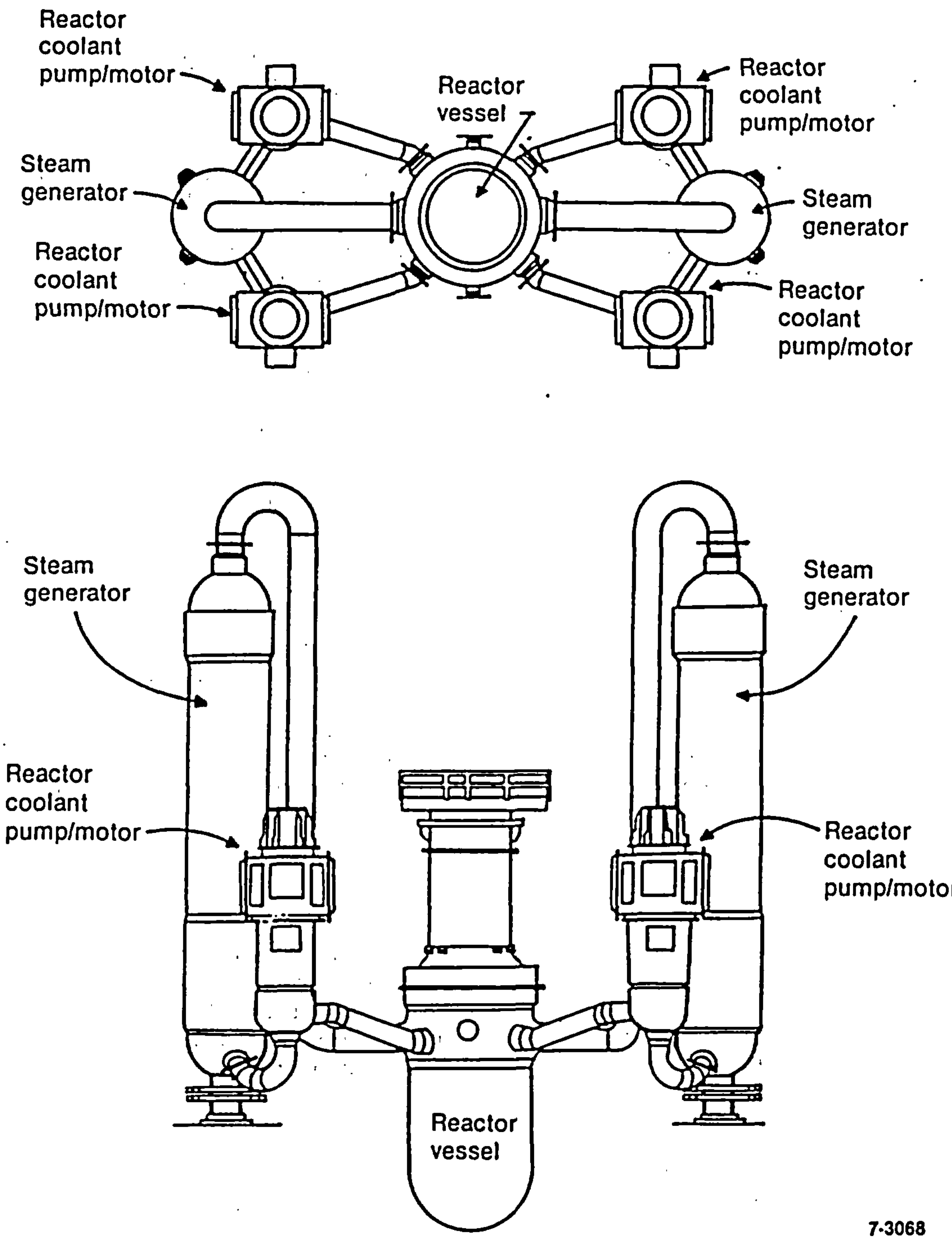

Figure 5.1. Babcock & Wilcox system.

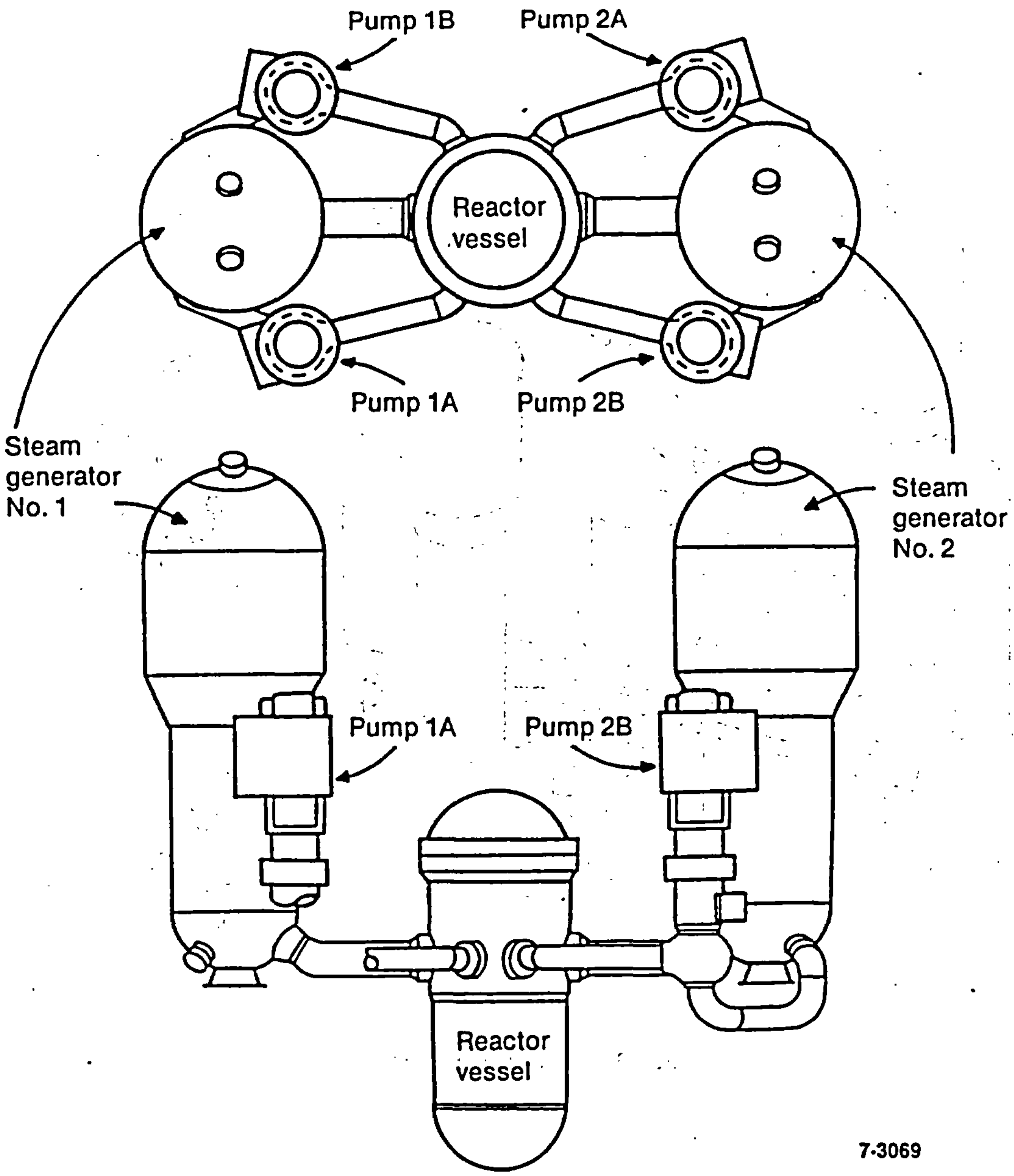

Figure 5.2. Combustion Engineering system.

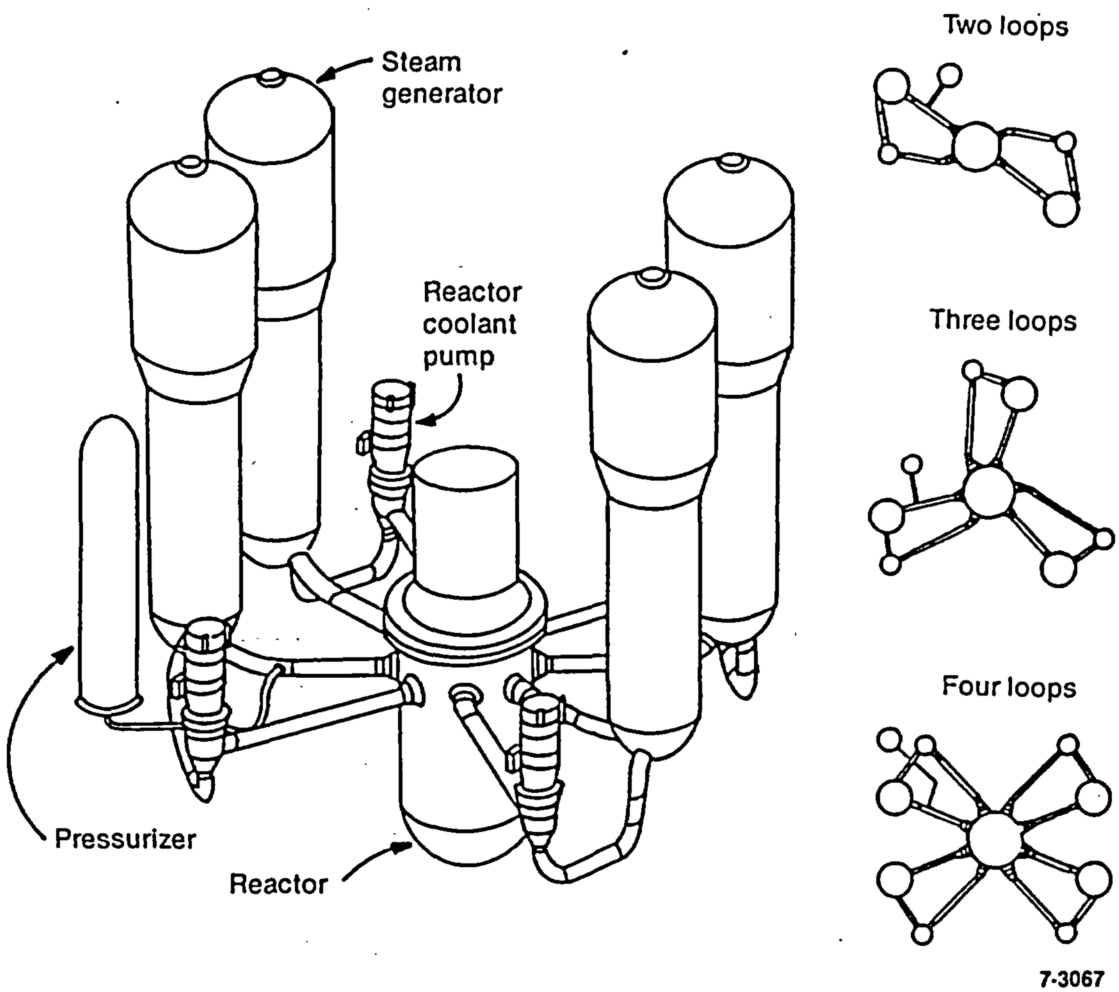

Figure 5.3. Westinghouse system.

Table 5.1. PWR main coolant pipe material

Vendor	Piping	Fittings	Cladding[a]
W[b]	CF8A[c] CF8M	CF8A	N.A.
CE	SA 516 GR 70	SA 516 GR 70 Type 309L SS	Type 308L SS
B&W	SA 106 GR C	SA 516 GR 70 Type 308L SS Type 309L SS	Type 304L SS

a. Welds between austenitic stainless steel (SS) and ferritic material normally have an Inconel 600 layer to overcome the need for sensitizing stress relief heat treatment.

b. Early W PWR plants had wrought Type 304 SS main coolant piping.

c. CF8A and CF8M are cast grades of Type 304 and Type 316 SS respectively.

systems and equipment is a major cause of transients. These transient conditions contribute additional thermal stresses to the main piping.

In addition, the plants are designed to be able to accept unusual conditions, such as severe earthquakes and dynamic effects resulting from catastrophic pipe failure in connected systems. These various conditions are discussed in the following paragraphs.

5.2.1 Steady-State Stress. All PWR primary systems are designed to freely expand as their temperature is increased from ambient to operating temperature, a rise of ∿500°F (280°C). In all cases, the reactor vessel is rigidly attached to the reactor building and serves as a fixed point when the system heats up and expands. The hot-leg piping attached to the reactor vessel nozzles expands along its axis to the steam generator. All three PWR designs have SGs that move freely to accommodate the pipe expansion.

The W SG is supported on columns with pinned connections that rotate freely and permit the SG to assume the natural or unconstrained hot position of the piping. The CE and B&W designs have SGs supported on skirts with sliding bases that accomplish the same result. The SGs for all three vendors move ∿2 in. (50 mm) during the plant heatup from ambient to operating temperature. The reactor coolant pumps also are supported in such a manner that cold-leg expansion is essentially unrestrained.

In the steady-state condition then, the state of stress in the main coolant piping is dominated by the internal pressure contribution. There are small amounts of thermal stress that result from secondary effects because perfect free expansion cannot be achieved. In addition, there are small amounts of gravity-induced stress that are inconsequential because of the short, stiff nature of the primary piping. Because the thermal expansion is unconstrained, there is little or no steady-state bending stress. The piping itself is of relatively large diameter with very short lengths.

For the reasons given above, the main coolant piping may be more accurately thought of as a series of pressure vessels than piping. This concept is important from the viewpoint of life extension, because the measures to be taken with main coolant piping will be the same measures to be taken with the connected pressure vessels, in contrast with those measures needed for long runs of relatively flexible piping.

As discussed above, the steady-state stress is essentially given by the pressure contribution. The stress is uniform tension in the pipe wall and is written as

$$S_h = \frac{pr}{t} \tag{5.1}$$

$$S_l = \frac{pr}{2t} \quad (5.2)$$

$$S_r = \frac{p}{2} \quad (5.3)$$

where

- p = pressure
- r = pipe mean radius
- t = pipe thickness
- S_h = circumferential or hoop stress
- S_l = longitudinal or axial stress
- S_r = radial stress.

The American Society of Mechanical Engineers (ASME) code stress intensity (P_m) is equal to twice the maximum shear stress, shown as

$$P_m = \frac{pr}{t} + \frac{p}{2} \quad (5.4)$$

and is limited to the ASME code design stress S_m.

S_m in turn is limited to the smaller of 2/3 S_y or 1/3 S_u

where

- S_u = specified minimum ultimate tensile strength
- S_y = specified minimum yield strength

and these quantities are the minimum specification values for the steel in question. Because of the high strain hardening nature of some austenitic stainless steels, S_m may be >2/3 S_y but never >90% of S_y nor 1/3 S_{ut}, where S_{yt} and S_{ut} are the yield and ultimate strengths, respectively, at temperature.

The foregoing paragraphs describe the steady-state stress on the pipe wall. As mentioned, there may be small amounts of additional stress caused by secondary constraints, but these are not significant. The stress picture is quite different, however, at discontinuities or changes in section. Stresses will be much higher at openings in the pipe wall where auxiliary pipe nozzles are attached and also at terminal ends where the pipe is welded into the primary vessels. In general, the elevated stresses at these locations are the result of purely geometrical effects that tend to produce incompatible deformations under the pressure load. The continuity of the structure requires compatibility, however, and results in locally higher stresses at geometrical discontinuities. As will be discussed later, these stresses are secondary in nature and are important mainly from the viewpoint of fatigue.

5.2.2 Transient Stresses. Each PWR primary system of the current generation is designed for a specific set of transient conditions. That is, the stresses resulting from a specific set of transients satisfy the limits of the ASME code. Earlier plants that did not fall under the requirements of Section III of the ASME code of 1971 or later or of American National Standards Institute B31.7 were not specifically required to be analyzed for transient stresses.

In this segment, nuclear plant transient conditions and the stresses resulting therefrom are discussed. As mentioned earlier, different generations of plants had to meet increasingly stringent analytical requirements, but as a practical matter the physical configurations of the plants changed but little, and the actual transient conditions were also very similar. Therefore, little or no generality is lost by discussing transient loads and stresses in current-generation plants.

5.2.2.1 Transient Conditions. The manufacturer of the PWR system is required by Section III of the ASME code to compile a design specification that lists all transient design conditions. The design specification contains a description of the transient conditions, usually in the form of a temperature-time relation, and lists the number of occurrences for which each transient condition must be considered. Design conditions and transients have historically been grouped into the following categories:

- Design
- Test
- Normal operation

- Upset conditions
- Emergency conditions
- Faulted conditions.

Faulted conditions normally include the safe shutdown earthquake and loss-of-coolant accidents. If a nuclear plant is afflicted by either of these events, there is no doubt that it will receive considerable special attention, probably far exceeding any life extension program procedures. In view of this consideration, and the extremely low probability of occurrence of faulted plant conditions, these conditions will be omitted from this discussion on life extension.

All the system transients in the remaining categories are considered in the plant analysis. Most of the transients are in the category of normal and upset conditions. To understand the effect of the transient condition, a specific transient, the system heatup and cooldown is discussed in some detail. This transient is especially appropriate for several reasons: it certainly must occur for the plant to operate, it is one of the simplest transients, and it contributes significant fatigue damage.

Ideally, in a plant designed for 40 years, the primary system would need to be shut down only during refueling, say conservatively once a year or on the order of 40 times during life. For design purposes, usually 200 (or even 500) heatup, cooldown cycles are specified to account for nonideal operation.

Normally the heatup would begin with the coolant at ambient temperature and pressure. In PWR plants, the heatup is controlled by a pressure-temperature relation that limits the pressure according to the temperature elevation. This pressure-temperature limit ensures that the reactor vessel will not be pressurized in a cold condition when the fracture toughness of the ferritic steel is depressed, particularly later in life after some radiation embrittlement has occurred. Heat is added to the system gradually, usually the heat loss from operation of the reactor coolant pumps and the temperature and pressure increase gradually, usually at a rate on the order of 50°F/h (28°C/h) although specification limits may be higher, say 100°F/h (38°C/h). Once the temperature is elevated sufficiently, the control rods can be pulled and the heatup completed with reactor heat. In a cooldown, the process is reversed.

Temperature gradients develop throughout the primary system in the course of the heatups. From a stress viewpoint, the most significant temperature gradients are those from the inside diameter to the outside diameter of the piping and nozzles. The warmer inside diameter seeks to expand and is restrained by the cooler outside diameter. The situation is exacerbated at nozzles and other locations of changing geometry and wall thickness. After steady state is reached, the gradients even out and the transient stresses vanish. Each change in system operation produces these effects in a greater or lesser degree. For example, once the system is at full power, a change in power level lowers the hot-leg temperature, which develops gradients in the pipe and associated transient stresses.

The significance of the transient stresses is that they, being of a cyclic nature, can cause fatigue. This is further discussed in Section 5.3.1. As can be seen from the foregoing discussion, the damage can be limited by reducing the number of transients and the rate of change of temperature.

5.3 Degradation Mechanisms and Sites

The primary-coolant piping consists of the hot-leg piping, cold-leg piping, pressurizer surge-line piping, and the several nozzles in the pipe connected to, for example, the charging line, safety-injection line, residual-heat-removal line, etc. In this section, various mechanisms are discussed that tend to reduce the life of piping and the nozzles in the general case. For the sake of completeness, an attempt has been made to discuss the applicability of most such mechanisms to the primary coolant piping of the PWR. No particular order is followed.

5.3.1 Fatigue. There is an entire list of mechanisms that can cause failure of pressure vessels and pipings and have done so in the past. As has been discussed in previous paragraphs, very few of these mechanisms are applicable to the main coolant piping of PWRs. However, one of the most widespread and best known, namely fatigue, is applicable. So far as is known, there have been no failures, not even discovered cracks, in existing PWR reactor coolant piping. That does not mean, however, that there will not be fatigue problems in the future.

The significant fatigue mechanism is low-cycle fatigue caused by a combination of pressure and transient thermal stresses. The points of highest stress throughout the systems are the most vulnerable. These sites include the nozzles welded into the primary-coolant piping and the terminal ends of the piping. Each transient event (that is, each heatup cycle, each test of the safety injection system, each operation of the charging pump, etc.) produces a cycle of stress throughout the system. At the geometrical irregularity of a nozzle or metallurgical irregularity of a dissimilar metal weld, these stress cycles may be significant. It is perhaps worthwhile to digress briefly to describe fatigue damage and the means of calculating it.

At a given point subject to a cyclic stress of sufficient magnitude, a crack will form. Then, during succeeding cycles the crack will grow until it progresses through the wall. It was shown by Coffin[1] that the fatigue life of a small, smooth-polished specimen was proportional to the range of strain imposed on the specimen in a cyclic constant strain range test. Since most of the fatigue life of a small smooth specimen is in the precrack phase, it may be inferred that the range of strain determines the onset of crack initiation.

Later it was shown by Paris[2] that the amount of crack growth of a given crack under a given cyclic stress correlated almost exactly with the applied fracture mechanics stress intensity range, ΔK, which is a function of the crack size, shape, stress level, and geometry. Detailed discussions of crack initiation and growth during low-cycle fatigue are provided in Section 3.4.2 of Chapter 3 and in Chapter 13 of this report.

The primary system piping is designed and analyzed on the basis of Section III of the ASME code. The ASME code provides a fatigue design curve derived directly from smooth, small specimen failure data by imposing safety factors of 2 on stress or 20 on cycles, whichever is lower. These factors are intended to account for size, environment, and other differences between laboratory specimens and actual equipment.[3] Large-scale vessel fatigue tests have been performed for the express purpose of checking the ASME fatigue design curves.[4] It was shown by these tests that cracks may be expected to initiate below the fatigue curve, but that wall penetration is not expected until fatigue usage exceeds the design curve.

The ASME code also employs Miner's linear damage rule to combine fatigue damage that accumulates at different stress levels. In this way, fatigue is accumulated according to a cumulative usage factor that is equal to unity when the accumulated fatigue usage is just equal to the ASME code design fatigue curve. See Section 3.4 for a detailed discussion on this topic.

Referring then to the tests of Reference 4, it may be inferred that for those locations in the primary system that have design fatigue usage factors near unity and in-service actually experience the design usage, then it would not be unreasonable to find fatigue cracks initiating near the end of design life. Of course, there are mitigating factors such as variable sequences of load cycles, etc. that could improve the situation. Nevertheless, it is clear that low-cycle fatigue cracking is definitely one of the most important damage mechanisms that will have to be dealt with in any life extension program for primary piping.

5.3.2 Thermal Aging. Cast austenitic-ferritic (duplex) stainless steels with significant amounts of delta ferrite will experience a reduction of toughness when aged at elevated temperatures. The maximum effect occurs at 885°F (475°C) and the general phenomenon is often identified as 885°F or 475°C embrittlement. This temperature is well above the maximum piping temperatures of PWR main coolant piping; nevertheless, the phenomenon obeys a time-temperature relationship such that at lower temperatures the toughness properties are still affected, but a longer time is required. The general phenomenon and the effect on PWR main-coolant piping is discussed at length in References 5, 6, and 7.

In the references cited, substantial investigations have been carried out and it has been suggested that in the design lifetime of existing PWR plants, the safety and integrity of the materials affected has not been compromised. The reason for this statement is that the cast stainless steels are very tough initially, and the amount of aging that occurs at the reduced temperatures of PWR operating systems does not lower the properties enough to render the materials susceptible to brittle fracture, nor to affect the crack-growth rates significantly to compromise the design capability.

It turns out that the degree of aging is related to the volume percentage of delta ferrite in the duplex material. It would definitely seem that piping or components such as pump or valve casings that have high delta ferrite content that are in the hot-leg side of the plant would become susceptible to brittle fracture at some period during an extended lifetime. These components will in the future require careful monitoring. Just as the reactor vessels are monitored today, both as to property changes and the appearance of cracks.

The NRC Office of Research is sponsoring a program at the Argonne National Laboratory to study the long-term aging effects on cast duplex stainless steels.[8] Service-aged, as well as laboratory-aged, specimens are being examined in this program. Close monitoring of this program is recommended.

Additional information on the thermal aging of cast stainless steel components may be found in Chapter 10, Boiling Water Reactor Recirculation Piping, of this report.

5.3.3 Stress-Corrosion Cracking. As is well known, intergranular stress-corrosion cracking (IGSCC) occurs in metals when tensile stress is present, the environment is conducive to stress corrosion, and the material has been sensitized. There are different conducive environments but the halogen salts promote stress corrosion cracking in austenitic stainless steels, and caustic environments promote stress-corrosion cracking in ferritic steels.

In the PWR primary piping, to date, there have been no known cases of stress corrosion cracking. Cracking is suppressed by good control of the environment including scavenging of oxygen to very low levels. Also, the boric acid used for reactivity control is believed to contribute to suppression of stress-corrosion cracking. In any event, unlike the boiling water reactor (BWR) plants, those few PWRs with wrought stainless steel piping have shown no evidence, so far, of susceptibility to stress-corrosion cracking.

Later W plants with main coolant piping of cast stainless have also been impervious to stress-corrosion cracking; however, this resistance to cracking was definitely expected. The cast stainless steel is a duplex alloy with significant percentages of delta ferrite that renders these alloys greatly resistant to stress-corrosion cracking.

The B&W and CE plants have ferritic steel piping and therefore are free from the threat of stress-corrosion cracking because there is little or no possibility of caustic or related compounds entering the primary system. Conceptually, the stainless-steel cladding would be vulnerable to cracking in some circumstances, but there has not been a problem; nor is it expected in the wrought stainless piping of the early W plants.

In summary, stress-corrosion cracking does not seem to be a problem in the primary piping system of any of the PWRs. A caveat on this conclusion, however, is that the piping should nevertheless continue to be inspected closely, since it is to be remembered that no stress corrosion was expected in the Inconel steam generator tubes, but after a few years it appeared. Similarly, for BWR recirculation piping, IGSCC was not expected but occurred.

5.3.4 Corrosion. Ordinary metal wastage because of any of the well-known electrochemical mechanisms of corrosion has never been a problem in the primary system piping of any of the PWRs. It is not expected to be in the future either. The main reasons for this good experience are that oxygen is completely removed from the system, and other impurities that could lead to electrochemical action are likewise removed. Primary PWR coolant is extremely pure water.

5.3.5 Brittle Fracture. Brittle fracture is the most feared of all mechanical-failure mechanisms because these types of failure are invariably sudden, complete, and catastrophic. This mode of failure is not considered possible in PWR primary piping because all the piping materials that have ever been used in this application are more than sufficiently tough. Indeed, recent calculations for all plants have proved that leaks will occur well before fracture is possible.

5.3.6 Erosion and Cavitation. Flow velocities in primary systems are well within acceptable limits for austenitic stainless steel, whether used for piping or cladding. As a result, erosion of main coolant piping has never been a problem, nor is it expected to be one in the future.

Cavitation erosion is not an uncommon problem in the process industries. In the PWR primary systems, however, there are operating restrictions to ensure net pump suction head limits are never exceeded. Similarly, the static overpressure is more than enough to control cavitation at other flow disturbances in the system.

5.3.7 Creep. In light-water nuclear plants, all system temperatures are below the creep range, i.e., less than 0.3 Tm, where Tm is the melting point of the material. Creep and other time-dependent deformations are insignificant.

5.3.8 Summary. In summary, the most significant degradation mechanism in the primary PWR piping is low-cycle fatigue. The locations at which this occurs first are the primary piping nozzles. In general, the larger-diameter nozzles tend to be most susceptible, as well as those nozzles with the most severe transients (such as the safety-injection nozzles). However, the true ranking of the nozzles is difficult to determine because the results are dependent on the type and exactness of the analyses (which is variable) and the actual transient usage, which is somewhat unknown. In addition, each manufacturer analyzes the system somewhat differently. The general results for all significant degradation sites are shown in Table 5.2.

5.4 Recommendations for Future Research

The fundamental and most significant damage parameter is fatigue because of system transient operation. However, nuclear plant operators rarely keep a careful accounting of the number, type, and severity of transients. The truth is that very few nuclear plants know with any certainty how much of their design life has been consumed at any given time.

Therefore, it is strongly recommended here that one or two plants, at least on a pilot basis, institute an in-depth study of past operations in order to gain some insight into the rate that the design life is being consumed. The issue sounds simple, but it is not. The plant was qualified to idealized descriptions of the transients. Actual plant transients do not conform to the idealization and are rarely as severe as the design descriptions.

The second recommendation is that programs be put in place now to monitor the actual degree of thermal-aging embrittlement. This issue is not seen as a current problem, but it could turn out to be a problem if very long-term operation were to be achieved.

Table 5.2. Summary of degradation processes for PWR primary system piping

Rank of Degradation Site	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Methods
1	Main coolant pipe nozzles[a]	System operating transients	Fatigue crack initiation and propagation	Through wall leakage	Volumetric inspection for diameter ≥4-in.
2	Terminal end dissimilar metal weld[b]	System operating transients	Fatigue crack initiation and propagation	Through wall leakage	Surface inspection for diameter <4-in.
3	Cast stainless steel	Temperature	Thermal embrittlement	Through wall leakage	Volumetric

a. The nozzles in the primary coolant piping are the highest ranking degradation sites. Additional specificity is avoided because the most severely loaded nozzles are difficult to determine. Reported results are heavily dependent on type of analysis performed. Actual damage will completely depend on the actual transient usage that occurs in the plant. This is the basis for the recommendation.

b. In W plants, the dissimilar metal welds are at the reactor vessel and steam generator nozzles.

5.5 References

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6. PRESSURIZED WATER REACTOR STEAM GENERATORS

V. Malhotra

6.1 Description

The steam generator is an important part of the pressurized water reactor (PWR) primary system pressure boundary. Failure of the steam generator tubes will provide a passage for the primary system coolant to the outside of the containment by way of the relief valves. The recirculating and the once-through steam generators shown in Figures 6.1 and 6.2, respectively, are the two types of steam generators commonly used. In the recirculating type steam generator, the secondary system water is fed into the downcomer (Figure 6.1) where it is mixed with recirculating water that is draining from the moisture separators. The downcomer water flows to the bottom of the steam generator, across the tube sheet, and then upward through the tube bundle where steam is generated. The primary system coolant flows through U-tubes. It enters the steam generator at 599 to 621°F (∿315 to 327°C) on one side and leaves at 550°F (∿288°C) on the other side.

A variation of the above recirculating design is the use of a section of the secondary side as a feedwater preheater to achieve greater thermal efficiency. In this, a section of tubes on the outlet or cold-leg side of the U-bends is partitioned off from the rest of the secondary side to form a preheat section. Water is then fed to the preheat section rather than into the downcomer.

In the once-through steam generator, the secondary-system water enters a feed annulus (Figure 6.2) above the ninth tube support plate level. Here it is mixed with steam aspirated from the tube bundle area and preheated to saturation. The saturated water flows down the annulus, across the lower tube sheet, and upwards into the tube bundle where it becomes steam. This superheated steam flows radially outward and then down the annulus to the steam outlet connection.

Operating experience has demonstrated that steam generators have had and will probably continue to have aging-degradation problems. These degradation problems are mainly associated with the recirculating type of unit that is manufactured by either the Westinghouse Electric Company or the Combustion Engineering Company. Some of these steam generators have developed thousands of degraded tubes after just seven or eight years of operation.[1] The five- to ten-year span of operation appears to be the critical timeframe in which many nuclear plant steam generators begin to exhibit severe tube degradation.

In the existing steam generators, the tubes are made of Inconel 600 with carbon steel as the support material. Over the years, various defects have caused tube failure as shown in Figure 6.3. A defective steam generator tube is defined as any tube which has been plugged for whatever reason. A large number of tubes have been plugged, however, the tubes that actually leaked are a small proportion of the tubes plugged. Phosphate wastage was the major cause of steam generator defects in the early 1970s.[2] From 1976 to 1979, denting was the major cause of tube failures in PWR steam generators. In 1980, stress corrosion cracking/intergranular attack (SCC/IGA) was the major cause of tube defects (31%), followed by denting (29%), and wastage (9%). In 1982, the major causes of tube defects were primary side SCC (54.5%), pitting corrosion (11.7%), and secondary side SCC/IGA (10.3%). The major causes of tube defects in 1983 and 1984, were primary side SCC and secondary side SCC/IGA.[3]

6.2 Stressors

Stressors in steam generators are any load or environment that tends to affect the functional capability of a part or component. Stressors include chlorides, copper compounds, air leakage into the system, type of water treatment, applied stresses during fabrication, tube support design, pH, temperature, velocities, and crevices. These factors, either in combination or separately, can produce intergranular stress corrosion cracking (IGSCC), SSC/IGA wastage, erosion-corrosion, pitting, fretting, or denting.

6.3 Degradation Sites

The type of degradation acting on steam generator tubes varies; however, it can be divided into the two broad categories of chemical and mechanical degradation. The following is a short description of the various chemical and mechanical degradation mechanisms and the sites at which they have been observed:

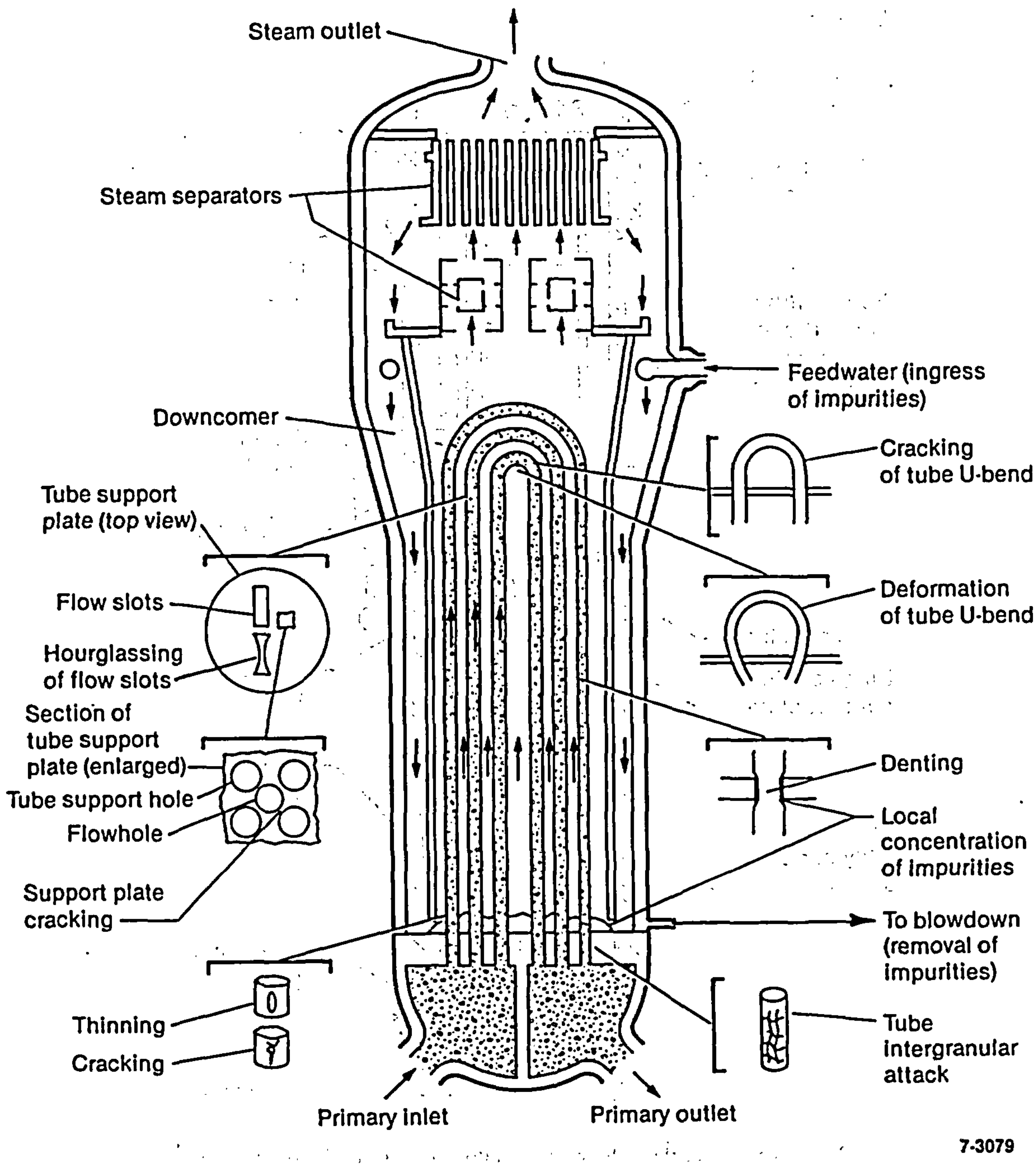

Figure 6.1. Sketch of recirculating steam generator with indicated problem areas.[4]

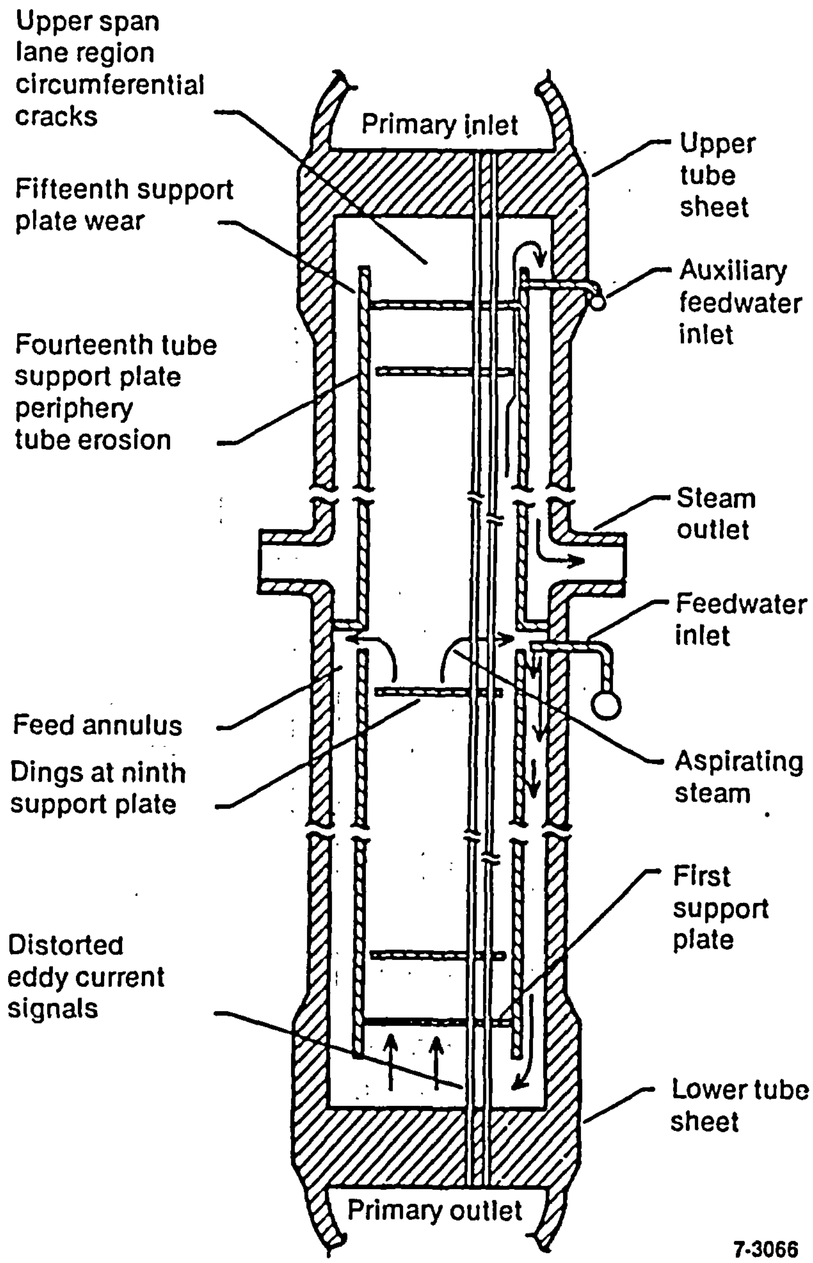

Figure 6.2. Sketch of once-through steam generator with indicated problem areas.[4]

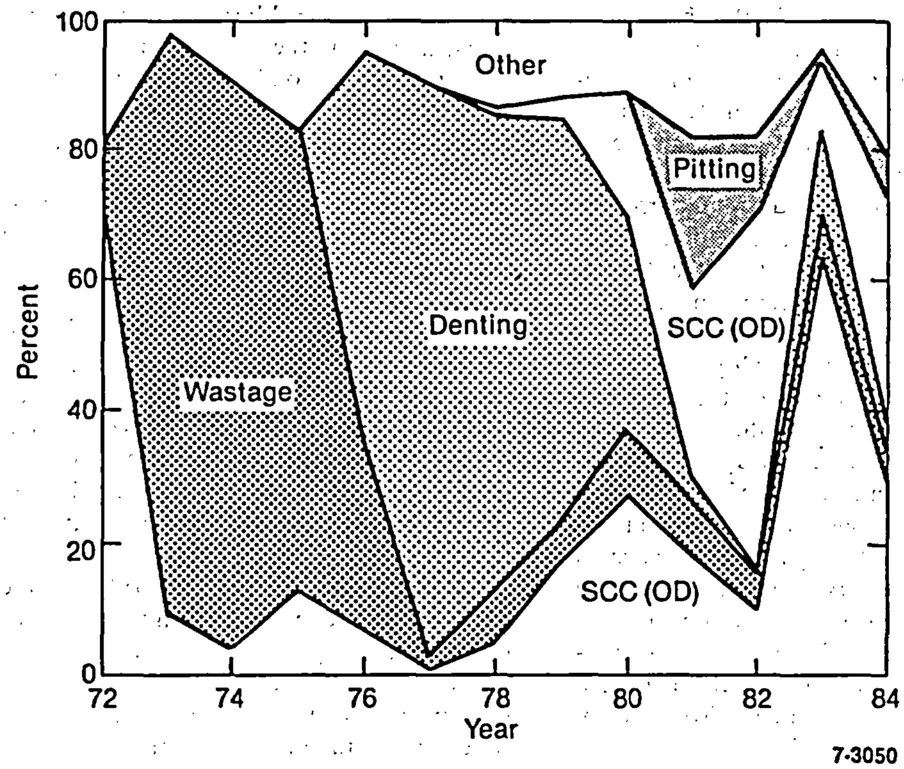

Figure 6.3. History of steam generator tube failure mechanisms.[3]

- Wastage (thinning): General corrosion of the outside surface of tubing in the region above the tube sheet when phosphate chemistry is used. Sludge provides a mechanism for the concentration of chemicals that attack the Inconel 600 tubing.
- Denting: The plastic deformation of tubes resulting from buildup of carbon steel support-plate corrosion products (magnetite) in the tube-to-tube support plate annuli.
- Pitting: A localized attack on the outside surface of tubing resulting from nonuniform corrosion rates caused by the formation of local cells. Pitting can be severe in the area of the sludge pile or where scale containing copper deposits is found. Pitting of Inconel 600 tubes takes place in the temperature range 212 to 392°F (100 to 200°C), hence, pitting is found mainly on the cold-leg side of the steam generator. However, Pacific Northwest Laboratories (PNL) has reported the presence of pitting on the hot-leg side in the original Surry steam generator.
- Intergranular Attack (IGA): Grain boundary attack on the outside surface of Inconel tubing with no stress orientation. This attack is noted in the hot-leg tube-to-tube sheet crevice region and tube-to-tube support plate crevice region.
- Intergranular Stress Corrosion Cracking (IGSCC): The primary side IGSCC sometimes initiates on the inside surface of the tubing at the U-bends of tubes in the inner rows with small bend radius and the roll-transition region where the tube is expanded into the tube sheet. The secondary side IGSCC initiates on the outside surface of the tubing in the crevice formed by the tube and the tube sheet, and tube and tube support plate.
- Fretting: The loss of tube material caused by excessive rubbing of tubes against their support structure. This can be caused by either primary side or secondary side flow-induced vibrations of the tubes. Severe fretting has occurred at baffles in the integral preheater near the feedwater inlet, and at antivibration bars in the U-bend region.

- Erosion-Corrosion: The mechanical damage caused by a corrosive environment and impingement of liquid or solid particles in a system. The carbon steel J-tubes in the early Westinghouse-designed steam generators have experienced wall thinning and perforation caused by an erosion-corrosion mechanism. The carbon steel J-tubes have been replaced by the J-tubes made of improved corrosion resistant materials to avoid this type of degradation, and therefore they will probably not have a significant impact on the life extension of these steam generators.
- Environmental Fatigue: This degradation is a result of synergistic effects of corrosion and fatigue. The locations of the degradation sites within the once-through steam generators include the secondary face of the upper tube sheet and the uppermost tube support plate near the open lane. The upper shell to transition cone girth welds in a recirculating steam generator have also been degraded by corrosion fatigue.
- Thermal Fatigue: The welds at the auxiliary feedwater nozzle thermal sleeve in once-through steam generators are subjected to thermal fatigue. The cracking of this weld will increase the potential of thermal shock to the steam generator shell.

6.4 Degradation Mechanisms and Failure Modes

Corrosion in recirculating steam generators tends to occur in stagnant regions, where impurities can concentrate and attack the tubes. These regions include the tube-to-tube sheet crevices, the tube support crevices, and the sludge zone above the tube sheet. The following sections describe the major types of degradation mechanisms and their failure modes.

6.4.1 Wastage. Wastage, or tube thinning, or corrosion of the tube material primarily takes place on the outside of the tubes in the area of a sludge pile. The sludge collects in the steam generator and causes chemicals to concentrate.[4,5] The result is that the tube wall thinning occurs in the region above the tube sheet when phosphate chemistry is used. It is believed that the rate of thinning, in part, is a function of the chloride and oxygen concentrations, as well as the sulfate concentration. Resin leakage from the condensate polisher bed is postulated to provide an acidic sulfate environment contaminated with chlorides. The progression of tube wall thinning appears to be a serious problem only in those plants that are still on phosphate water chemistry control. About 200 tubes were plugged in 1982 because of the wastage problem. Closer control of the phosphate water treatments was attempted by a number of utilities in order to avoid steam generator tube wastage and cracking, but these efforts were unsuccessful. Thus in order to halt these problems, many utilities changed the secondary water treatment from phosphate to ammonia. Hydrazine chemistry [all-volatile treatment (AVT)], in essence, eliminates phosphate problems.

6.4.2 Denting. Oxygen, chloride, and copper oxide and their compounds can form a voluminous (nonprotective magnetite) corrosion product on the carbon steel support plate and subsequently deform some of the steam generator heat transfer tubes at the tube-to-tube support plate intersections. This reduction in tube diameter was subsequently termed denting. The forces that cause the indentation result from the accelerated corrosion of the carbon steel support material because of chloride contamination of the steam generator water. Chloride ingress from condenser leaks concentrate at the tube/support structure intersection and lead to fast linear growth of nonprotective magnetite.

Several nuclear power plants experienced steam generator tube-denting problems in the early 1970s. The incidents of denting were observed both in plants that were originally on phosphate chemistry control and then switched to an AVT and plants that were on AVT from start-up. However, the plants experiencing tube denting generally used sea water or brackish water for condenser cooling and were therefore susceptible to higher and more acidic levels of chloride contamination from condenser inleakage than were plants that used fresh water. Some steam generator tube denting has been observed in recent years in plants that are on AVT control and fresh cooling water; however, only plants with sea water cooling have shown extensive tube denting to date.

Research has been undertaken to understand the role of certain parameters such as chemistry, temperature, heat flux, crevice condition, and materials in the tube-denting mechanism, and to determine the threshold concentration levels at which denting occurs. Devices such as the isothermal capsule, single-tube-model boiler, heated-crevice apparatus, and inleakage concentrator were used in these studies. Additionally, existing data from various operating plants were analyzed to understand the tube-denting mechanism.[5] A summary of some of the conclusions derived from the above work follows.

An acidic chloride environment is required at the crevice surface of the carbon steel support for the occurrence of denting or nonprotective magnetite formation (NPM). The threshold concentration of strongly acid-forming chloride solutions (copper chloride or hydrogen chloride) for the formation of NPM is between 175 and 700 ppm chloride. Sufficient NPM was produced in solutions with concentrations of 700 to 1500 ppm chloride to cause tube denting. The threshold concentration of a weak acid-forming chloride solution for the formation of NPM is also between 175 and 700 ppm chloride. The threshold chloride concentration in sea water for NPM formation is between 2000 and 10,000 ppm and for tube denting is 20,000 and 90,000 ppm chloride. Oxygen in the makeup water made a sea water environment more aggressive with respect to the production of NPM.

In heated-crevice tests with prepacked crevices in a 1% sea water (200 ppm chloride) and copper oxide environment, a crevice superheat of only 0.3°F (0.17°C), which yields a concentration of 2000 ppm chloride (based on boiling point elevation) produced significant denting. The denting rate increased with initial crevice superheat in the range of 0 to 1°F (0.56°C). Above this, the denting rate decreased with increasing superheat up to a value of 23°F (12.8°C). As the size of an initial open crevice was reduced in these tests, a width of 4 mils was reached for isothermal capsules and 2 mils for heated crevice tests at which tube denting is reduced or stopped.

The rate of tube denting increased in isothermal capsules containing 5800 ppm chloride as the temperature increased from 450 to 600°F (232°C to 316°C). Also, when the pH of the chloride solution was above 4, NPM was not formed even with chloride levels of 35,000 ppm chloride.

Cooling tower water containing 0.34 ppm chloride and silicon dioxide, which is mildly alkaline-forming when heated to steam generator temperature, produced denting when copper oxide was added as an oxidant. Silicon dioxide by itself produced slight denting in a nondenting environment.

Increasing the chromium content of steel decreased the tendency for NPM formation. Type 405 stainless steel in place of carbon steel did not produce NPM. Therefore the support plates in the new steam generators are made of Type 405 stainless steel. Quatrefoil designs for support plates are currently used to reduce further the damage due to denting.

Large quantities of hydrazine (40 ppm) in the blowdown were not immediately effective in stopping denting in either a copper chloride or an oxygen environment.

Corrosion rates are a function of available superheat, but they also depended on the concentration of chloride present. The existing plant sludge data revealed that plants with a copper-to-iron ratio greater than 1 developed accelerated denting, while those with a copper-to-iron ratio less than 1 had little or no indication of denting. Current trends are for sea-water- and brackish-water-cooled plants to use AVT with condensate demineralization and titanium instead of copper as the condenser tube material.[3]

6.4.3 Pitting. Pitting is an extremely localized attack that results in holes in the metal (pitting occurs because of a localized breakdown in passivity). In most cases these holes are relatively small with a surface diameter about the same or less than the depth. It is difficult to detect pits because of their small size and because they are often covered with corrosion products. Pitting usually requires several months to years to show up. Once initiated, however, it progresses at an accelerating rate, and failures often occur with extreme suddenness. For this reason, it is difficult to detect pitting and the best fix for life extension is to eliminate the conditions that lead to pitting.

With the exceptions of Millstone-2 and Indian Point-3, only shallow pitting has been observed in PWR steam generator tubes removed from service. The maximum pit depths measured in tubes removed were ~0.005-in. Pitting was sufficiently minor that these pits were not detected by eddy-current testing nor did the existence of these pits promote concern about possible primary-to-secondary leakage.

As a result of the extensive pitting in the Millstone-2 and Indian Point-3 steam generators (40% throughwall pits), the Electric Power Research Institute (EPRI) conducted a workshop[6] on steam generator tube pitting. It was reported that both affected plants had pits of similar morphology (chromium oxide and copper metal-filled undercut pits). These pits were found under scale and in similar locations (e.g., the cold-leg side near the sludge pile). PNL has also reported the presence of pitting in the hot-leg side in the original steam generator removed from the Surry plant.

It has been shown in the laboratory that it is possible to pit Inconel 600 under a variety of conditions but only a few of the laboratory-grown pits have the same morphology that is found in operating plants. Electricite de France has produced pits in low-oxygen, high-chloride (740 ppb), high-sulfate (760 ppb) water. These pits had chromium-rich, copper-included scabs similar to those found in the Millstone and Indian Point steam generators. There is no universal agreement as to whether steam generator pit initiation and growth is primarily an operating temperature (hot) phenomenon or a layup (cold) phenomenon. However, chromium oxide-filled pits are seen in high-temperature tests, but not in low-temperature tests. Also, laboratory tests indicated that the chloride content of water and pitting are closely related, and that pits grow under sludge or scale and are associated with copper deposition. PNL reports that oxygen inleakage will also cause pitting of steam generator tubes. There is no straightforward explanation about why pits generally appear on the cold-leg side and what role, if any, resin ingress may play in the initiation and growth of pits. Also, no mention is made of the effect of pitting on corrosion fatigue in the literature.

6.4.4 Intergranular Attack. IGA has been observed in Inconel 600 tubing in the hot leg and tube-to-tube sheet region of some of the operating steam generators. From analytical determinations, it appears that for IGA to occur, a highly alkaline condition must exist in the crevice caused by the concentration of alkaline species present in the secondary side water. It was noted that a combination of IGA and SCC is often present close to each other in failed tube samples, with the IGA being more extensive than the SCC. Tubes removed from existing plants[7] indicate that both the outside diameter surface and the intergranular fracture face had in addition to the three major elements (nickel, chromium, and iron), the presence of sodium, potassium, calcium, phosphorus, sulfur, aluminum, and chloride.

Tests were conducted using high-temperature electrochemical measurements to identify conditions leading to this kind of IGA.[8] The results of these tests indicated that in 10% caustic media at 608°F (320°C), IGA is commonly observed in Inconel 600 in the mill-annealed condition. Thermally-treated material at 1292°F (700°C) showed definite improvement over mill-annealed material in both resistance to IGA and SCC.

The presence of various anions strongly influences the corrosion attack. Carbonates and sulfates, and to a lesser extent phosphates, are very deleterious and can develop deep IGA and SCC, depending on the value of the electrochemical potential. The electrochemical potential of the Inconel 600 tubes is governed by the composition of the secondary water during operation. When AVT control is used, reductive conditions are encountered, and in the case of caustic pollution, IGA should preferentially occur in crevice regions. On the other hand, when oxygen enters the steam generator (either in the form of oxygen or metallic oxides), the potential is raised and favors an SCC mechanism. For this reason, the composition of the sludge and, in particular, the oxidizing potential could be the deciding factor in whether IGA or SSC will occur.

Occurrence of SCC/IGA at or within the tube sheet can be treated by sleeving affected tubes. The large-scale sleeving at Point Beach-2 and San Onofre-1 included many tubes with SCC/IGA. The possibility of SCC/IGA at the tube sheet may also be reduced by expanding the tubes for the full depth of the tube sheet and, thus, eliminating the crevices.

Secondary side SCC/IGA has also been found, recently, in the crevice formed by the tubes and tube support plates.[3] This degradation mechanism at the support plates caused several failures in 1983. The SCC/IGA at the support-plate intersections cannot be eliminated by closing the crevices and it is of greater concern because it can occur at more than one level in the steam generator.

6.4.5 Intergranular Stress Corrosion Cracking. Increasingly, tubes have been removed from service as a result of primary-to-secondary leaks at points of high stress concentration: the Row 1 U-bends, the rolled transition regions in the tube sheet, and dented intersections. These cracks appear to have initiated on the inside surfaces of the tubing. It is not clear whether inside initiated cracking of nondented units is a sporadic occurrence related to some flaw in the manufacturing process, which imparts excessive stresses to that surface, or if it is a generic problem[4] characterized by a long initiation time and slow rate of progression or both. Inside surface initiation of tube cracking is believed to be related to pure water or the Corion phenomenon. In the first Corion tests, pure water was shown to crack Inconel 600 at temperatures well above the primary side operating conditions. Laboratories are trying to define quantitative relationships between failure time and variables such as strain rate, stress, environment, and temperature.

The outside surface initiation of SCC in the secondary side of the tube-to-tube sheet crevice region was discussed in Section 6.4.4.

6.4.6 Fretting. Tube fretting and wear is caused by vibration. Tube vibration can be induced by fluid crossflow perpendicular to the tubes or by parallel flow along the tubes.[4] In the PWR steam generators, a clearance exists between the tube and the tube support plate through which the tube passes. This clearance is required for manufacturing and design considerations. Vibration of the tubes can cause impact and tangential sliding of the tube against the tube support plate and adjacent tubes, resulting in local wear.

Fretting regions are sensitive to fatigue cracks. Under fretting conditions, fatigue cracks can initiate close to the material surface at very low stresses, well below the fatigue limit of nonfretted tube material. If the vibration cyclic stresses in the tube are sufficient to cause these fretting-induced cracks to grow and propagate, early failure of the tube can occur.

Fretting has caused extensive tube damage at the antivibration bars in several steam generators. This damage has been blamed on the poor design of the antivibration bars that were in point contact with tubes. The design has now been corrected, and the new design provides a broader region of contact with the tubes.

6.4.7 Erosion-Corrosion. Impingement of liquid or solid particles is known to cause mechanical damage,[4] especially to protective surface films. This can happen to a variety of materials for certain impinging velocities, sizes, shapes, and hardness of particles and impinging angles. When the erosion factor is combined with a corrosive environment, the mechanical damage can be accelerated.

The impingement of solid particles on steam generator tubes under operating conditions can cause mechanical damage. The probable sequence of events is that solid particles entrained in the bulk flow impact against the tube, promoting erosion. In a noncorrosive environment, the erosion process removes metal from the tube wall. In a corrosive environment, the erosion process may first remove a protective film from the tube, thus making the tube susceptible to corrosion. In both cases, wall metal loss occurs, either directly or by accelerated corrosion of the tube surface. Sources of solid particles can include any particles entering the steam generator and those particles entrained from within the steam generators.

Erosion-corrosion has occurred in once-through steam generators around the fourteenth support plate. This process has caused more than 40% wall reduction in some steam generators.

6.4.8 Environmental Fatigue. Some once-through steam generators, particularly early units, have an open lane, i.e., an untubed lane in the steam generators.[9] The steam flow carries entrained water droplets up the open lane to the upper tube sheet region. Evaporation of water concentrates any contaminants in these droplets and also carries the chemicals up the open lane to the upper tube sheet region. Here the droplets impact the heat transfer tubes and dryout, thus further concentrating the chemicals on the steam generator tubes. Even chemicals that are generally innocuous in the feedwater can cause metal loss in high concentrations.

Tube samples removed from alongside the untubed lane revealed a serpentine band of shallow metal loss—sometimes containing microcracks on the tube outside surface just below the secondary face of the upper tube sheet or near the highest tube support plate. Several tube failures, attributed to fatigue, are associated with this band of metal loss. Circumferential transgranular cracks have also led to primary to secondary leaks. The serpentine metal loss acts as sites for fatigue crack initiation, since the steam generator tubes do undergo cyclic vibration.

Laboratory tests have indicated that an acid solution containing sulfates, silicates, and chlorides can produce metal loss and microscopic indications (microcracks) closely resembling those observed from field-examined tubes.

6.5 In-service Inspection Methods

A host of in-service equipment has been designed for steam generator inspection, some of which have been used on existing power plants. This equipment provides information about tube denting, intergranular stress corrosion cracking, pitting, tube-to-tube support gap, etc. A brief description of some of the techniques available to the industry follows:

- Hydrogen Evolution Monitoring as a Measure of Steam Generator Corrosion.[10] High-temperature aqueous corrosion of carbon steel in steam generators produces hydrogen and magnetite. Measuring the hydrogen content of the exiting steam is, at present, the only available on-line means of monitoring steam generator corrosion. It also can provide information about the effectiveness of the corrosion inhibitors and the effect of changes in water chemistry. There is no available information as to whether or not this technique has been used at any operating plant.

- Eddy-Current Nondestructive Examination for IGA. Routine inspection of steam generator tubing is conventionally performed using a bobbin-type eddy-current coil.[11] However, IGA defects have not been detected successfully in all the cases. Extensive copper deposits on the secondary side of the U-bends and the hot-leg side of the tubes interferes with the eddy current inspection. Successful detection has been achieved by the use of pancake coils that are arrayed with their axes parallel to the tube radius and multiplexed to permit use of available electronic packages. Field eddy-current data show that volumetric intergranular attack can be detected when the flaws are from 20 to 30% through the wall, whereas an oriented IGA crack can only be detected when the flaws are about 60% through the wall. Oriented IGA flaws tend to be oriented along the axial direction within the tube sheet crevice region, and along the circumferential direction at dented tube sheet intersections and in the roll transition regions. (Additional discussions on the eddy-current examination technology are given in Section 11.2.4 in Chapter 11 and Section 14.1.1 in Chapter 14 in this report.)
- EMAT System. This method addresses ultrasonic inspection of steam generator tubing using an electromagnetic acoustic transducer (EMAT).[12] It was developed to detect flaws in certain areas of tubes where conventional single-frequency differential-coil eddy-current has had difficulty, including circumferential cracks, defects at dents or support plates, and defects in U-bends.

  The EMAT system has a good defect-detection capability for circumferential cracks and other defects that provide a fairly wide and sharp circumferential-oriented cross section (e.g., dented areas); however, the system has difficulty detecting flaws with small cross sections, i.e., cracks that are tight. For example, the system can inspect U-bends with its advance signal but does not effectively detect the axial cracks that are found in U-bends. In addition, the system has limited defect depth-sizing capability.
- Optical Probe for Steam Generator Tube Dent Measurement.[13] The optical profilometer has been tested successfully under laboratory conditions. It can measure inside tubing profiles in the range of radii from 0.32 to 0.40 in., with an average calibration error of 0.005 to 0.008 in. on nominal inner diameter and +0.006 in. on dents. Preliminary investigations suggest that the calibration error is most probably caused by variation in interior surface finish.
- Pulse-Echo Ultrasound for Steam Generator Tube-to-Support Plate Gap Measurement.[14] This equipment uses an ultrasonic technique for determining the condition of the gap between steam generator tubes and support plates, which allows monitoring corrosion product buildup that might lead to denting. It also can determine the efficiency of chemical cleaning for removing this buildup.

Other nondestructive methods include sonic leak detectors and helium leak testing to search for air leakage into the system:

- Sonic leak detectors are used to check all of the potential leak locations during the period after a vacuum has been drawn on the condenser but before the unit gets to load. This technique is used regularly in finding steam leaks in boilers and has been used on industrial vacuum systems. Its use on power plant vacuum systems is new.
- Helium leak testing is used to search for air leaks after the unit reaches partial load. A load of at least 20% is necessary for efficient leak testing.

6.6 Summary, Conclusions, and Recommendations

The major causes of PWR steam generator tubing failures were identified in this chapter along with the probable causes of the identified failure modes. Conclusions were drawn, where applicable, at the end of the discussion of each failure mode.

The degradation mechanisms for the steam generator tubes are prioritized based on operating experience as summarized in Table 6.1. For example, in 1982, IGSCC was the number one cause of tube defects, followed by IGA, and so on. To control or mitigate each failure mode, the following recommendations are made:

1. Wastage. Wall thinning of tubes has been reported in plants that use phosphate water chemistry. The number of tubes affected by phosphate water chemistry is small; this problem can be controlled by periodic lancing of sludge to minimize tube degradation, affected tubes may be sleeved, or the plant changed to AVT control.
2. Denting. Denting is no longer an important cause of tube failures. In 1982, only 1.4% of the defective tubes were dented, mainly because of more stringent control of the secondary side chemistry by reactor operators. Some of the measures that have helped to control denting in older steam generators include sludge lancing, minimizing condenser and air leaks, eliminating copper alloys from the feed train, using boric acid, installing deep bed polishers, and reducing the oxygen in the condensate storage tank. The steam generators should be subjected to the soak procedure to remove as much chloride as possible. In this method, the steam generators are soaked at about 300°F (149°C) for at least 8 h, with maximum bleed and feed, and then the steam generators are cooled to below 200°F (93°C), drained under a nitrogen blanket and refilled with clean water. Newer steam generators that are better designed and equipped with more corrosion-resistant tube supports made of ferritic stainless steel (SS405) should not suffer denting if good water chemistry is maintained.
3. Pitting. Two reactors (Indian Point-3 and Millstone-2) reported severe secondary side pitting of their Inconel 600 tubing. The pitting occurred in the cold leg between the tube sheet and the first support plate. Ingress of brackish water, air, and copper is believed to have assisted the pitting process. To mitigate pitting, remove all copper material, minimize chloride and air ingress, change to AVT control, and maintain strict chemistry control.
4. Intergranular Stress Corrosion Cracking. The only demonstrated method to prevent inside-surface-initiated IGSCC in an over-rolled tube is to peen the inside surface. This places all of the inside surface in compression where parts of it after over-rolling are in tension and therefore subject to IGSCC. A Westinghouse method called Rotopeening has not been satisfactory in steam generators that have been operated; however, it has been successful in new steam generators. Shot-peening has been highly successful in both new and previously operated steam generators.

   The cracks resulting from inside-tubing-surface-initiated stress corrosion, caused by over-rolling, are difficult and in some cases impossible to detect by nondestructive examination of the installed tubes because the tube sheet provides interfering signals. If cracking does occur, the tubes should be sleeved or their lower ends replaced.

   IGSCC has also been reported in Row 1 of the U-bends in several steam generators. Development of an in situ stress-relief procedure is recommended to mitigate such cracking. Testing has shown that thermally treated Inconel 600 provides better resistance to IGSCC than annealed Inconel 600. If leaks do develop in the U-bends, they should be plugged. The use of thermally treated Inconel 690 for tubing for new steam generators should be considered because the available metallurgical and corrosion data show that it is the most corrosion-resistant Inconel tube alloy.[15] The French are already using this alloy for their steam generator tubings. Kraftwerkunion AG uses Inconel 800 for steam generator tubing and shot-peens it after bending to introduce compressive stresses in the U-bend that eliminate stress corrosion cracking.

   On the secondary side, SCC/IGA are widespread problems that affected 14 reactors in 1982. This problem led to replacement of the steam generators at Point Beach-1 and large-scale sleeving at Point Beach-2. Tube degradation occurred within the deep tube-to-tube sheet crevice in 11 reactors. The short-term solution to SCC/IGA is to install sleeves of either thermally treated Inconel 600

Table 6.1. Summary of degradation processes for steam generator tubes

Rank[a]	Degradation Site	Stressors	Degradation Mechanisms	Failure Modes	ISI Method
1	Inside surface of U-bends and roll-transition regions	Tube rolling stresses, Corion phenomenon, denting	IGSCC	Cracking	Eddy-current testing
2	Outside surface of hot-leg tubes in the tube-to-tube sheet crevice region	Alkaline environment, presence of SO_4 and CO_3 anions	IGA, IGSCC	May eventually result in cracking	Eddy-current testing
3	Cold-leg side in sludge pile or where scale containing copper deposits is found	Brackish water, air, and copper	Pitting	Local attack and tube thinning may eventually lead to a hole	Eddy-current testing, optical scanner system, sonic leak detector system
4	Outside surface of tubing above tube sheet	Phosphate chemistry, chloride concentration, resin leakage from condensate polisher bed	Wastage (thinning)	Uniform attack, tube thinning may eventually wear out the material	Eddy-current testing
5	Tubes in the tube-support regions	Oxygen, copper oxide, chloride, temperature, pH, crevice conditions	Denting	Flow blockage in tubes caused by plastic deformation	Helium leak and sonic leak testing, optical probes, hydrogen evaluation monitoring, pulse echo ultrasound method
6	Contact points between tube and antivibration bar	Flow-induced vibrations	Fretting	Wear out of material caused by rubbing and/or fatigue	Eddy-current testing
7[b]	Once-through steam generator tubes	Velocities, sizes, shapes, impact angle, and hardness of particles	Erosion-corrosion from impingement of particles	Wear out of material	Eddy-current testing
8[b]	Once-through steam generator tubes in the upper tube sheet region	Chemicals, (localized corrosion) vibrations	Fatigue	Primary to secondary leaks	Eddy-current testing

a. Based on operating experience for steam generator defects.

b. Denotes once-through steam generator (Items 7 and 8 do not reflect rank order). First six items are for recirculating steam generators.

or 690. A change to AVT control chemistry also is recommended.

5. Fretting. Seven reactors in 1982 experienced fretting damage to their steam generator tubes, caused by flow-induced vibration. In two cases, severe fretting occurred at baffles in the integral preheater near the feedwater inlet. Remedial actions included installation of modified impingement baffles and plugging of affected tubes. Fretting caused by antivibration bar movement in the U-bend region was observed in other units, which is considered a design problem in the older

units. In newer units, the problem is eliminated by reducing the gap size between the antivibration bars and the tubes.

6. Erosion-Corrosion. Affected tubes should be plugged. Stringent control on water chemistry should also be applied.
7. Environmental Fatigue. Most of the previously afflicted plants were partially bypassing their polisher during and shortly before most failures occurred. Action was taken to reduce the vibrational loads on the tubes, and to provide strict control of water chemistry (including restoration of 100% condensate demineralization) to prevent ingress or buildup of aggressive chemicals.

   Blockage of the open lane with a lane-flow blocker is used now. This forces the open lane's steam-water mixture out into the tube bundle at a lower elevation thus, dispersing the damaging chemicals before they can concentrate at a weak point in the upper tube sheet location.

To mitigate the various degradation mechanisms associated with steam generators, the manufacturers have taken or recommended the following steps:

- Change from drilled tube support plates to other designs, such as, the quatrefoil design in the Westinghouse Model F unit and the trifoil design for B&W steam generators. Additionally, shifting from carbon steel tube-support material to 12% chromium ferritic stainless steel.
- Strict control on water chemistry on the secondary side and a change from copper to titanium condenser tube material. Westinghouse now recommends mostly the AVT chemistry over phosphate treatment.
- To control IGSCC, heat-treated Inconel 600 or 690 now is used instead of annealed Inconel 600.
- The use of shot peening has been highly successful in the U-bend and rolled areas of the new and operating steam generators. The fretting problem is eliminated in the new steam generators by redesigning the antivibration bars. The new design has reduced the gap size between the tubes and the antivibration bars.

There are several issues associated with aging and life extension of steam generators that require followup. Life-assessment models are needed to determine the degradation of steam generator tubes by the two potentially most tstatus of corrosion within nuclear steam generator tubes. The significance damaging mechanisms: IGSCC and IGA. Monitoring methods are needed to determine of the changes made in the designs and material selection of the tubes and tube support plates needs to be evaluated. Also, useful results are expected from the EPRI program that is developing improved nondestructive-evaluation methods for steam generators, the USNRC steam generator program, and similar programs in France, Germany, and Japan.

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7. REACTOR PRESSURE VESSEL SUPPORTS FOR PRESSURIZED WATER REACTORS AND BOILING WATER REACTORS

W. G. Hopkins

7.1 Review of Types of Supports, the Materials and Environment

There are five major types of reactor pressure vessel supports used in light water reactor (LWR) plants: neutron shield tanks, columns, cantilever supports, brackets, and skirt type supports as shown in the Figures 7.1 to 7.5, respectively. The first four types of supports are used to support pressurized water reactor (PWR) vessels, while the fifth type is used to support boiling water reactor (BWR) vessels. The design responsibility for the four PWR types of support varies from vendor to architect/engineer, while the design responsibility of the skirt support lies solely with the reactor vendor because it is an integral part of the RPV. The major degradation mechanism for the RPV supports is irradiation embrittlement caused by neutron irradiation. Therefore, the designs of the RPV supports are discussed with respect to the susceptibility of their materials, e.g., ferritic steel and weld metal, to irradiation embrittlement. A detailed discussion of the damage due to irradiation embrittlement was presented in the Chapter 3 of this report and therefore, is not repeated here.

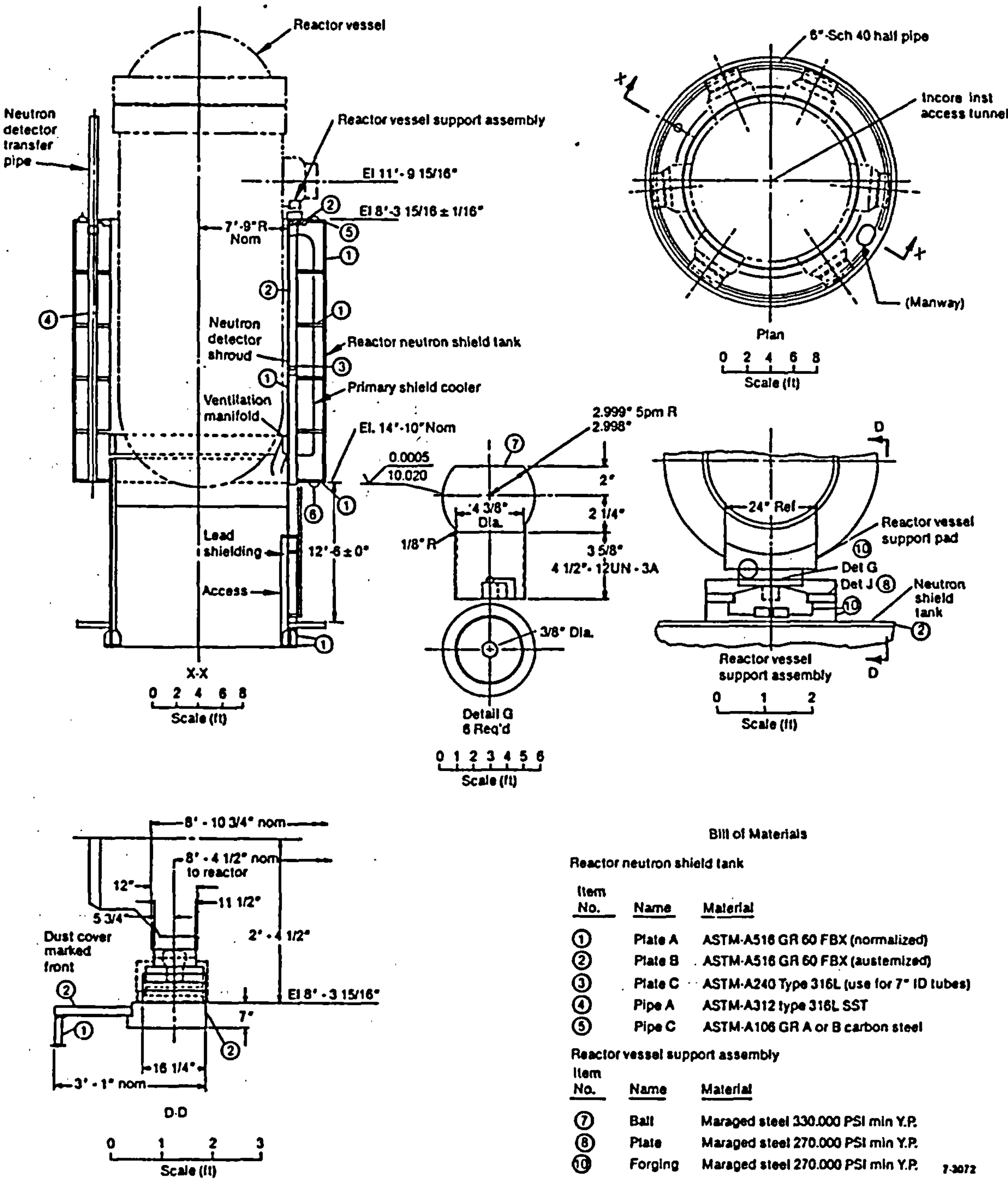
A major factor in producing nil-ductility-transition-temperature (NDTT) shifts in support materials is direct exposure to the reactor core beltline neutron flux. The neutron shield tank is a skirt-mounted, annular, water-filled tank formed by two concentric shells extending above and below the reactor core (as shown in Figure 7.1), and is fully exposed to the reactor core beltline neutron flux. Water is circulated through the tank for cooling and shielding. Similarly, the column type support is also fully exposed to the reactor core beltline neutron flux. The cantilever type supports may or may not experience significant neutron exposure, depending on a design's specific location relative to the core's location. The bracket type supports are sufficiently elevated above the region of significant flux and are, therefore, not likely to experience great shifts in their NDTT temperatures. The skirt supports used in BWRs do not experience significant shifts in their NDTT because the BWR reactor vessels are larger in size than the PWR vessels and have much more water between the reactor core and pressure vessel that absorbs neutrons and shields the pressure vessel, skirt and other hardware outside the vessel. However, the BWR skirt supports are subjected to fatigue damage when the RPVs change temperature and pressure during startup and shutdown.

In order for crack propagation to occur, the support member must be in tension at some point. With the neutron shield tank, column and skirt type supports, tension is experienced only under dynamic loadings, such as safe-shutdown-earthquake and loss-of-coolant-accident (LOCA) loadings. The bracket and cantilever supports experience tensile forces under their deadweight loads.

The North Anna units 3 and 4 have neutron shield tanks. In 1979, Virginia Electric and Power Company notified the USNRC (in accordance with 10 CFR 21) that predicted end-of-life NDTT shifts would exceed the operating temperature of the supports. The USNRC commissioned the Naval Research Laboratory (NRL) to perform a confirmatory study.[1] In that study, the NRL used displacements per atom (dpa) to predict NDTT shifts. This parameter was chosen so that embrittlement or hardening data taken from one neutron spectrum could be applied to the same material in a different spectrum. The end-of-life level (reactor life of 30 full-power years) of damage in dpa was calculated for the shield tank. The NDTT shift was calculated based on the assumption that the elevation in transition temperature is proportional to the square root of dpa. From this result, it was found that the estimated end-of-life NDTT exceeded the service temperature of the shield tank. In addition, it was predicted that the Charpy V-notch upper shelf energy level would be reduced to 30 ft-lb (41 J).

From this study, NRL recommended to the USNRC that additional investigations were warranted in the following areas[1]:

> "1. Determination of the range of radiation environment conditions (neutron spectra and flux levels) for other shield tank support structures and for support structures of the open column design (column support). Comparison of the findings with those for North Anna Units 3 and 4, relative to potential for notch ductility reduction.



Bill of Materials

Reactor neutron shield tank

Item No.	Name	Material
1	Plate A	ASTM-A516 GR 60 FBX (normalized)
2	Plate B	ASTM-A516 GR 60 FBX (austemized)
3	Plate C	ASTM-A240 Type 316L (use for 7" ID tubes)
4	Pipe A	ASTM-A312 type 316L SST
5	Pipe C	ASTM-A106 GR A or B carbon steel

Reactor vessel support assembly

Item No.	Name	Material
7	Ball	Maraged steel 330.000 PSI min Y.P.
8	Plate	Maraged steel 270.000 PSI min Y.P.
10	Forging	Maraged steel 270.000 PSI min Y.P.

Figure 7.1. Neutron shield tank RPV support.

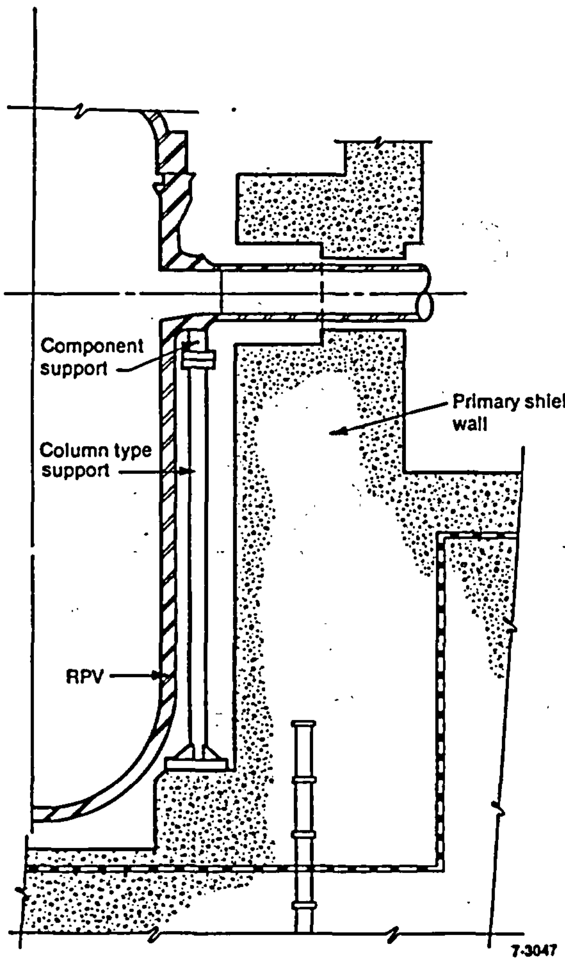


Figure 7.2. Column RPV support.

"2. Development of more detailed information on the upper shelf notch toughness of support structure materials in the transverse (weak) test orientation in the as-fabricated condition.

"3. Development of fracture toughness and strength assessments for low upper shelf steels irradiated at <450°F and a correlation of fracture toughness with Charpy-V properties in this condition.

"4. Assessments of the significance of radiation reductions in notch ductility and elevation in yield strength to support structure fracture safe reliability.

"5. Develop, for the steels of application, more detailed information on the potential for and sources of variable steel sensitivity to <450°F irradiation.

"6. If possible, conduct experimental irradiations, which will directly compare radiation effects of spectra in far-core versus near-core irradiation locations to verify the usefulness of the dpa approach for making radiation damage projections."

If the above recommendations by NRL are followed, the estimate for the shift in NDTT will be improved. An improved estimate for the NDTT shift will most probably result in increased margins for the plant life extensions.

Table 7.1[2,3] lists the steels most commonly used in RPV supports. The NDTT shifts are a function of irradiation temperature and neutron fluence (E >1.0 MeV). At lower irradiation temperatures [<450°F (232°C)] the NDTT shifts are more pronounced because of the inability of the material to self-anneal. In addition, those steels with relatively high copper contents will experience greater transition temperature shifts. Table 7.2[4] shows the computed relationship between copper content and irradiation temperature on the expected NDTT shift. The effect of neutron fluence levels at a constant irradiation temperature on the expected NDTT shift can be seen from Table 7.3.

The typical range of service temperatures for the RPV supports is 120 to 250°F (49 to 121°C). The temperature can fall to values as low as 50°F (28°C) when the reactor is shut down. This low temperature, combined with the neutron environment and copper content of the support material, can lead to significant shifts in NDTT. In addition, it has been found that when using dpa as a neutron exposure parameter rather than fluence, (in both cases E >1.0 MeV), greater NDTT shifts are calculated (see Reference 5, Table 1). Comparison with test results suggests that dpa might be a better neutron exposure parameter to use.[5] Table 7.4 is taken from examples presented in Reference 5 and illustrates the effect of copper content and fluence versus dpa calculations of NDTT.

7.2 Review of the Design Basis for RPV Supports

In considering plant-life extension (PLE) for the RPV supports a brief look must first be taken at their design basis. The General Design Criteria of 10 CFR 50 address the minimum design requirements with regard to licensing that must be met for structures, systems, and components important to safety. In particular, Criterion 2 states that the design of a component should be able to withstand

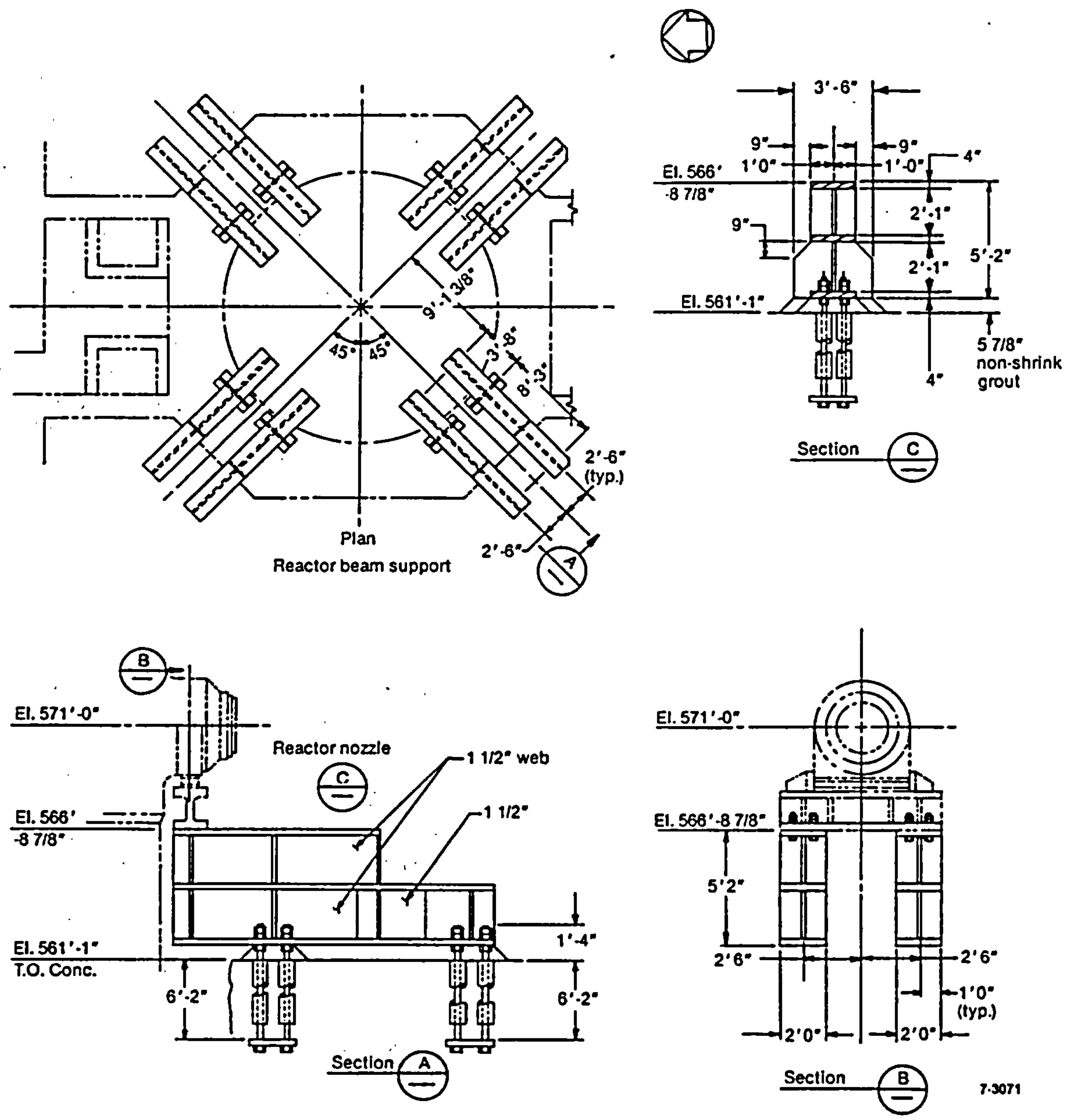


Figure 7.3. Cantilever RPV support.

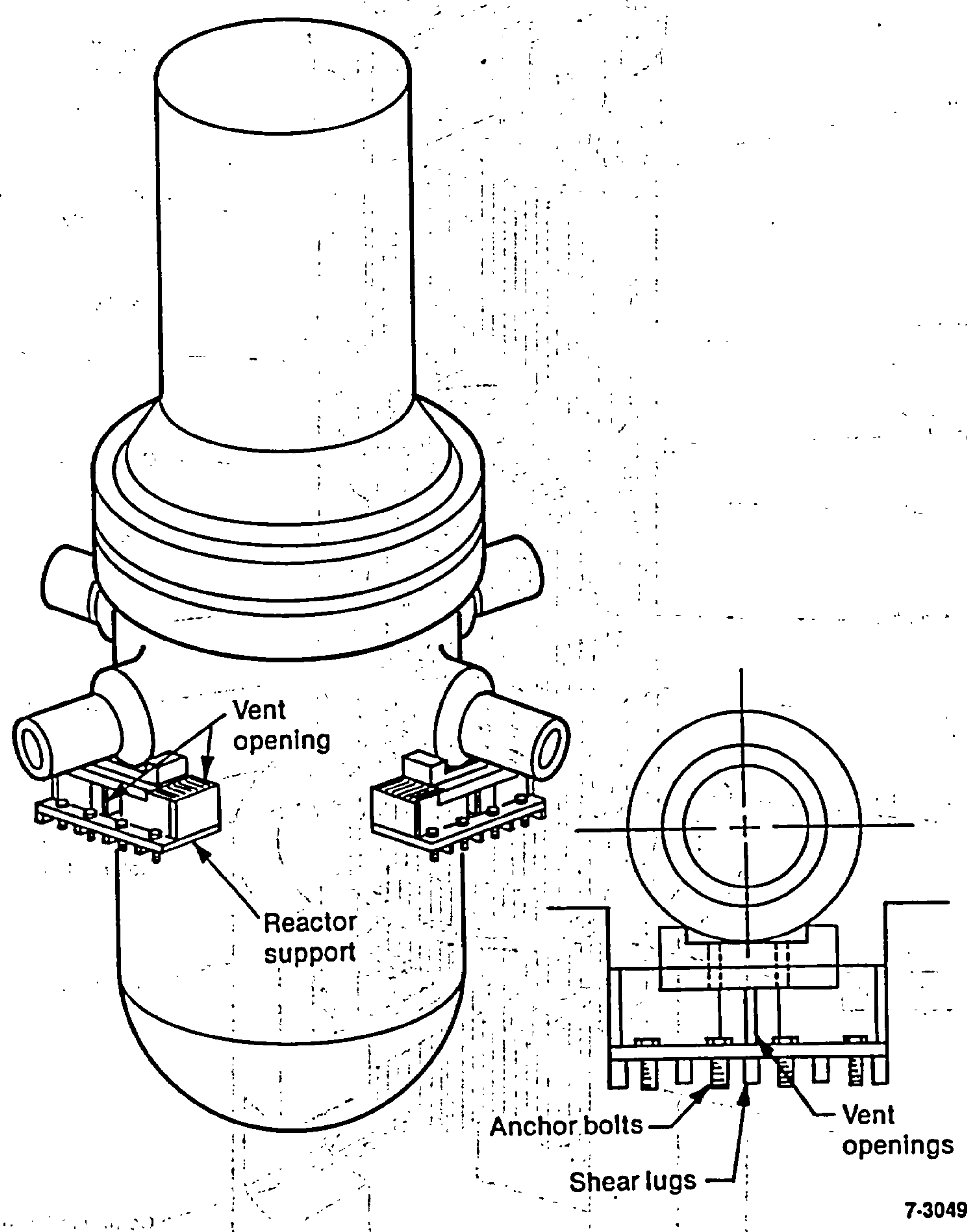


Figure 7.4. Bracket RPV support.

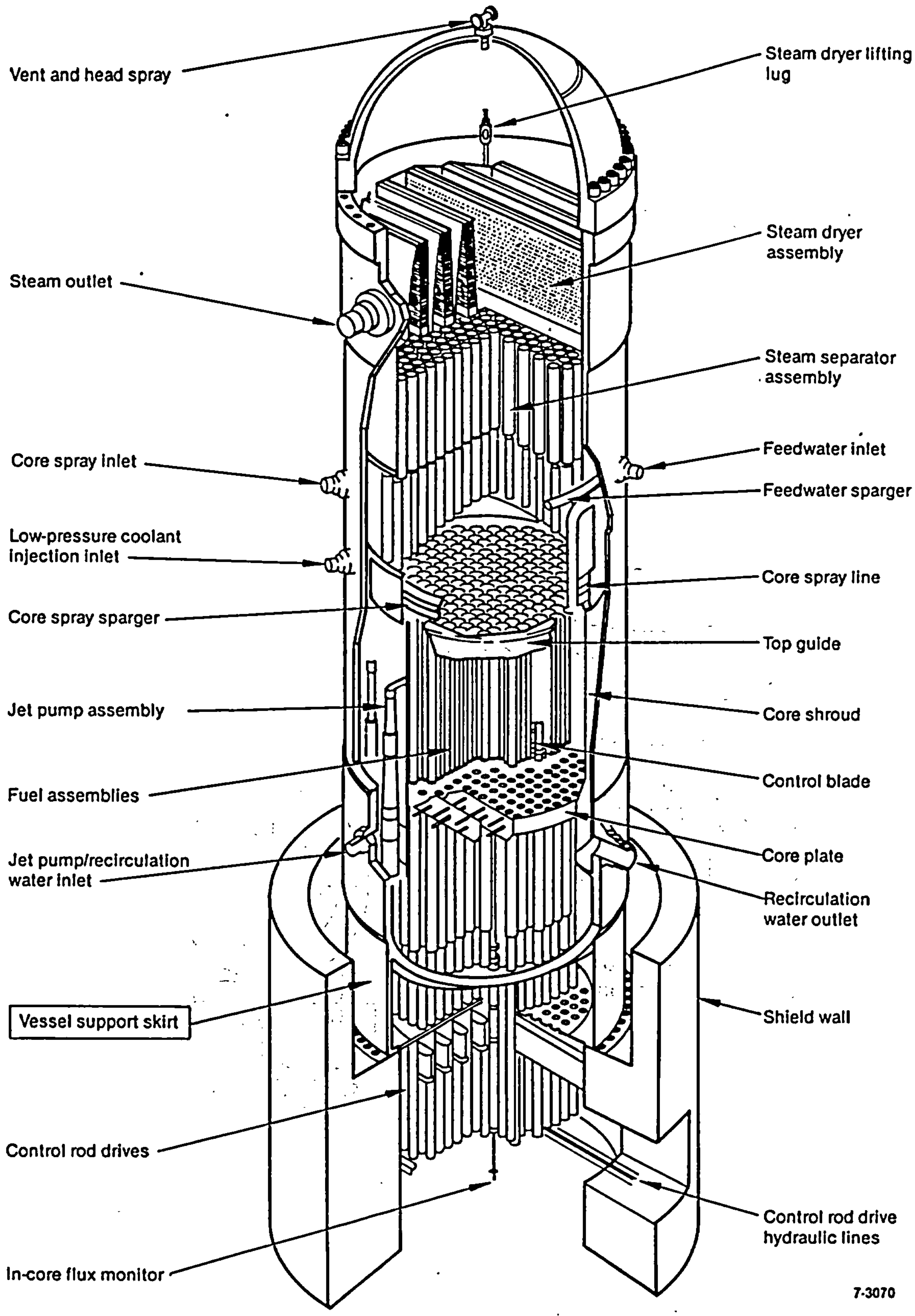


Figure 7.5. Skirt RPV support.

Table 7.1 Common RPV support steel characteristics 2,3,[a,b]

Type of Steel	Classification	Initial NDTT (°F)	Irradiation[c] NDTT Shift (°F)	Minimum Yield Strength (ksi)
A572 (normalized)	High-strength low alloy	-50	85	40
A302-B	Low alloy (not quenched and tempered)	0	85	50
SA533-B1	Quenched and tempered	0	85	50
SA516-70	Carbon-manganese	10	100	38
A572 (as-hot rolled)	High-strength low alloy	25	85	40
A36	Carbon-manganese	60-100	100	36

a. Private communication with P. B. Lindsay, Bechtel National, November 4, 1974.

b. Private communication with G. A. McCoy, Bechtel Power Corporation, January 28, 1976.

c. Fast fluence (E > 1 MeV) = 1.23×10^{18} n/cm^2.

Table 7.2. NDTT shifts at a neutron fluence of 10^{18} n/cm^2 (E > 1 MeV) for various copper levels, with 0.012% maximum phosphorus[4]

Copper (wt%)	550°F Irradiation NDTT Shift (°F)	450°F Irradiation NDTT Shift (°F)
0.35	105	158
0.25	75	113
0.20	58	87
0.10	25	38

Table 7.3. Dependency of NDTT changes of A302B steel upon various neutron fluence levels[a]

NDTT Increase (°F)	Neutron Fluence at 250°F (n/cm2 > 1 MeV)
80	1×10^{18}
85	1.25×10^{18}
100	2×10^{18}
150	4.1×10^{18}
170	5×10^{18}

a. Private communication with P. B. Lindsay, Bechtel National, November 4, 1974.

Table 7.4. Computed NDTT at PWR supports based on neutron fluence (E > 1 MeV) and dpa (E > 1 MeV)[5]

Fluence at Core Midplane[a] (E > 1 MeV) (neutrons/cm²) (A)	dpa at Core Midplane (B)	Copper Content (%)	NDTT °F (A)	NDTT °F (B)
1.08×10^{18}	9.03×10^{-4}	0.4	110	127
4.86×10^{17}	3.96×10^{-4}	0.1	23	26
3.08×10^{17}	2.92×10^{-4}	0.05	11	13

a. Assumed 40-year plant life and 80% capacity factor.

the most-severe-case effects of natural phenomena recorded for the plant site and not lose the capability to perform its safety function. In addition, the effects of natural phenomena must be appropriately combined with the effects of normal and accident conditions. Criterion 4 deals with the environmental- and missile-design bases. It states that the design of a component should "accommodate and be compatible with the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents." Also, "the components should be appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit."

In any combined event, the design-basis limitation and primary-failure mode is catastrophic brittle failure of the RPV support. Catastrophic brittle failure is a result of three conditions being met simultaneously. These conditions are (a) that a flaw of critical size is present, (b) that there is sufficient load on the support member to develop a critical stress level at the crack tip of the flaw and, (c) that the service temperature is low enough to promote a cleavage fracture at the crack tip of the flaw.[6] The NRL has developed a method that correlates the above needed conditions for catastrophic brittle failure. The method is known as the fracture analysis diagram (FAD)[7] and is based on a generalized stress-temperature diagram for fracture initiation and crack arrest.

Use of the FAD to determine fracture-safe operation requires determination of the NDTT, which can be accomplished by either Charpy V-notch tests or by drop-weight tests. The NDTT is defined as the highest temperature at which the stress needed for fracture initiation becomes the same as the stress taken from the curve representing the yield strength of the steel. The nominal stress needed for fracture initiation decreases in the presence of a flaw. As the flaw increases in size, the nominal stress required decreases. The FAD predicts that, for a given level of stress experienced above the NDTT, fracture initiation is possible only with flaw sizes equal to or greater than a critical flaw size. The critical flaw size is proportional to the square of the stress intensity K_{IC}. The crack-arrest temperature (CAT) curve relates the temperature to the applied nominal stress at which propagation of brittle fracture is arrested. For stress levels at one half the yield strength of a material and at temperatures 30 to 35°F above the established NDTT, it was found that there was a definite cutoff in fracture propagation. The CAT was, therefore, found to be directly relatable to the NDTT. Thus, the FAD method can be used to determine fracture-safe operation based on the NDTT of the material and expected stress levels. If the service temperature of the material in use never falls below the NDTT plus some safety margin temperature, then catastrophic brittle failure cannot occur. Should the service temperature fall below the NDTT plus safety margin under certain loading conditions, the determination of the critical crack size and location is required for the analysis, using linear elastic fracture mechanics,[8] to ensure continued safe operation.

Regarding the stress levels for RPV supports, the various loading combinations used in the design of RPV supports should be reviewed. These loading combinations are an explicit interpretation of the previously discussed NRC General Design Criteria 2 and 4 of 10 CFR 50 as implemented in detail by Section III of the American Society of Mechanical Engineers (ASME) code.[9] Table 7.5[10] shows the various loading combinations that must be accounted for under the four designated design conditions. Table 7.6[10] lists the appropriate parts of Section III of the ASME code for obtaining the appropriate stress limits allowed for each design condition.

With regard to brittle failure, special attention must be directed to those design conditions when the service temperature is at its lowest, i.e., during shutdown. If the applied loads, such as deadweight and safe shutdown earthquake, are high enough when combined with the low temperature, then two conditions for brittle failure are present.

Also of particular interest are the loadings associated with a postulated LOCA. It was discovered in May of 1975 that asymmetric loadings on the reactor vessel supports because of a postulated double-ended rupture of the reactor-coolant piping had not been taken into consideration in the original design of North Anna units 1 and 2 RPV supports. These asymmetric loadings result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel caused by transient differential pressures in the reactor cavity.[11] Because of these newly identified loads, the USNRC on January 20, 1978,[12] required that all PWR plants evaluate the adequacy of their reactor system components and supports under the effects of asymmetric LOCA loadings. Analysis of postulated guillotine breaks has shown that the additional loads experienced by the supports are approximately equal to the combined loads of deadweight and SSE.

Table 7.5. Design loading combinations for supports for ASME Code Class 1, 2, and 3 components[10]

Design Conditions	Design Loading Combinations[a]
Normal	DW
Upset	DW + OBE
	DW + RVC
	DW + FV
	DW + OBE + RVO
	DW + DU
Emergency	DW + DE
Faulted	DW + SSE + RVO
	DW + SSE
	DW + DF

a. Legend:
- DW Piping dead weight
- OBE Operating basis earthquake (inertia portion)
- SSE Safe shutdown earthquake (inertia portion)
- FV Fast valve closure
- RVO Relief valve - open system (sustained)
- DU Other transient dynamic events associated with the upset plant condition
- DF Dynamic events defined as a faulted condition

In addition to the above environmental parameters that influence the fracture toughness of RPV supports and their design lifetime, one other important environmental parameter must also be considered. This parameter is the degradation effects of neutron radiation embrittlement on the RPV support material discussed above and in Chapter 3. Only the RPV and the RPV support structures in a nuclear power plant experience this unique type of degradation mechanism. Consequently, Section III of the ASME Boiler and Pressure Vessel Code, Part NF-2121, requires that the materials used in reactor pressure vessel supports "... shall be made of materials that are not injuriously affected by ... irradiation conditions to which the item will be subjected." A recent report by the Electric Power Research Institute[13] has shown that there are significant fast neutron fluences (i.e., $>10^{18}$ neutrons/cm^2 at energies greater than >1 MeV) in the annulus between the outer surface of the reactor pressure vessel and the inner surface of the primary shield, in the region opposite the reactor core. As discussed previously, supports are located within this region in some pressure-vessel-support system designs and are therefore subject to shifts in their NDTT and subsequent decrease in fracture toughness. In addition, supports in this region are usually maintained at or below 150°F (66°C), so that the concrete in the primary shield meets the concrete code[14] requirements for maximum temperatures. Therefore, any inherent annealing in the supports is nil compared to the annealing in the reactor pressure vessel that is maintained at temperatures of ~550°F (288°C).

Studies[15] at the NRL have demonstrated the importance of reducing the copper content in steels to suppress both the magnitude of the NDTT shift and the decrease in upper-level fracture toughness. The NRL also has developed the drop-weight test to ascertain the true NDTT. The USNRC has incorporated the results

Table 7.6. Allowable stress limits for Class 1 component supports[10]

Support Type	Conditions				
	Design	Normal	Upset	Emergency	Faulted
Plate and shell design by analysis	NF-3221	NF-3222	NF-3223	NF-3224	NF-3225
Linear type supports by analysis	NF-3231	NF-3231	NF-3231	NF-3231	NF-3231
Component standard supports design by analysis	NF-3240	NF-3240	NF-3240	NF-3240	NF-3240
Component supports design by load rating	NF-3260	NF-3260	NF-3260	NF-3260	NF-3260

of this research into Regulatory Guide 1.99, Revision 2[16] for prediction of radiation damage in reactor pressure vessels.

With the emergence of the North Anna syndrome[17] and the additional loads it imposes on reactor vessel support systems, it is important that any decrease in fracture toughness and increase in NDTT in reactor pressure vessel supports be assessed in PLE considerations.

Regulatory Guide 1.99, Revision 1,[4] was used to calculate the increase in NDTT and assesses the potential lifetime of a RPV support in the following example.

1. Determine the actual current NDTT of the RPV support material by way of sampling and monitoring techniques as described below in Section 3 or by way of analytical estimates as described in the following steps.
2. Assess the fast neutron fluence to which the RPV support material will be exposed over its normal design basis lifetime (e.g., 10^{18} neutrons/cm^2 over a 40-year lifetime with an 80% plant capacity factor. This is equivalent to a fast neutron flux of 10^9 neutron/cm^2-s). This can be done using reactor physics calculations as outlined in American Society for Testing and Materials (ASTM) Standard E 482.
3. Determine the lowest temperature the RPV support material will experience [e.g., 50°F (28°C)]. This temperature is usually set in a technical specification in the license of the facility.
4. Determine the initial NDTT of the RPV support material from the material mill certification records [e.g. -23°F (-21.5°C)].
5. Determine the maximum loading conditions at the lowest service temperature. This establishes what safety margin must be used based on Pellini's FAD procedures, e.g., at half the yield strength, the FAD indicates that the material must be at NDTT + 30°F (NDTT + 17°C). For this example, it is assumed that the location of the maximum tensile stress (half of yield strength) in the support coincides with the location of maximum fluence. Then, for this example, the maximum NDTT shift must be given by

$$\underbrace{50°\text{F}}_{\text{(lowest operating temperature)}} - \underbrace{30°\text{F}}_{\text{(FAD safety margin)}} - \underbrace{(-23°\text{F})}_{\text{(initial NDTT)}} = \underbrace{43°\text{F}}_{\text{(maximum NDTT shift)}}$$

6. Using Regulatory Guide 1.99 (see Table 7.4), assess the NDTT shift of 25°F (14°C) for the normal operating lifetime for the support material's weight percent copper. However, because the supports are irradiated below 450°F (232°C), the 25°F (14°C) shift must be increased by 50%, (i.e., the supports in the example with 0.10% copper and 0.012% phosphorus, and a fast neutron fluence of 10^{18} n/cm^2 have experienced an NDTT shift of 38°F (20°C) over their 40-year lifetime).
7. The available PLE margin for temperatures less than 450°F, can now be determined in the following manner:

$$\text{NDTT margin for PLE} = \text{NDTT}_{\text{maximum}} - \text{NDTT}_{40\text{ years}}$$

$$\text{NDTT margin for PLE} = 43°\text{F} - 38°\text{F} = 5°\text{F}$$

8. Referring to Figure 7.6, a 3°F (2°C) increase [5°F (3°C), decreased by 50% for irradiation at temperatures greater than 450°F] in NDTT would result in an increase of the fast neutron fluence from the 40 year value of $\sim 1.0 \times 10^{18}$ neutrons/cm^2 to a fast fluence of 1.2×10^{18} neutron/cm^2. Recalling from Step 2 that the neutron flux was 10^9 neutrons/cm^2-s for our example, the PLE margin is given by

$$\text{PLE margin} = \frac{1.2 \times 10^{18} - 1 \times 10^{18}}{10^9} = \frac{\Delta(\phi t)}{\phi} = 2.0 \times 10^8 \text{ seconds}$$

where

ϕ = neutron fast flux

ϕt = neutron fast fluence

PLE margin = 6.3 years.

A reexamination of design-basis loads can also result in a similar PLE gain. This is because of a recent exemption granted by the USNRC when considering loads from post-LOCA guillotine breaks. Recent advances in fracture mechanics has shown that there is an extremely low probability of a double-ended break occurring in PWR primary coolant loop piping. Instead, it has been found that the PWR primary coolant pipe is much more likely to leak before it breaks, so that detection of small flaws by either in-service inspection or leakage monitoring systems is assured. This has led to a modification of General Design Criterion 4 Requirements by the USNRC. A statement has been added as follows:

> "... the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in pressurized water reactors may be excluded from the design basis when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions."[18]

The analysis referred to above is outlined in NUREG-1061, Volume 3, dated November 1984.[19] Once an extremely low probability (10^{-6} per reactor year) of rupturing is ascertained, redesign of the RPV supports is possible. In particular, support loadings because of the dynamic effects of postulated pipe breaks can be eliminated. This result, in effect, extends the lifetime of the RPV supports because the remaining loads (dead weight + earthquake) are about one half of the total design basis loads caused by the asymmetric loads from a LOCA. Thus, if a support steel had been chosen with a yield strength of 100 ksi with 50 ksi ensured for NDTT + 30°F (NDTT + 17°C), dropping the loading from 50 ksi to 25 ksi implies that a larger fast neutron fluence may be tolerated. In the previous example, where the final NDTT must be 43°F (6°C) because of the NDTT + 30°F (NDTT + 17°C), a new design basis can be used of NDTT + 15°F (NDTT + 9°C) (for one-quarter yield strength from the FAD) or the final NDTT is 50 - 15°F - (-23°F) = 58°F (14.5°C). Compared to the old value of 43°F (6°C), an additional 15°F (9°C) PLE margin is thereby realized. Again using the above example, with a 10^{18} fast neutron fluence at 40 years, this PLE margin of 15°F (9°C) can be translated from Regulatory Guide 1.99 as before into an extended lifetime. The 15°F (9°C) is only worth 10°F (5.5°C) [recall the 50% penalty for <450°F (<232°C) irradiation]; therefore, a 25°F (14°C) NDTT [38°F (20°C), for T <450°F (232°C)] corresponding to a starting fluence of 10^{18} would result in a final NDTT of 35°F (1.7°C). This corresponds to fast fluence from Figure 7.6 of 2×10^{18} neutron/cm^2. Proceeding as before

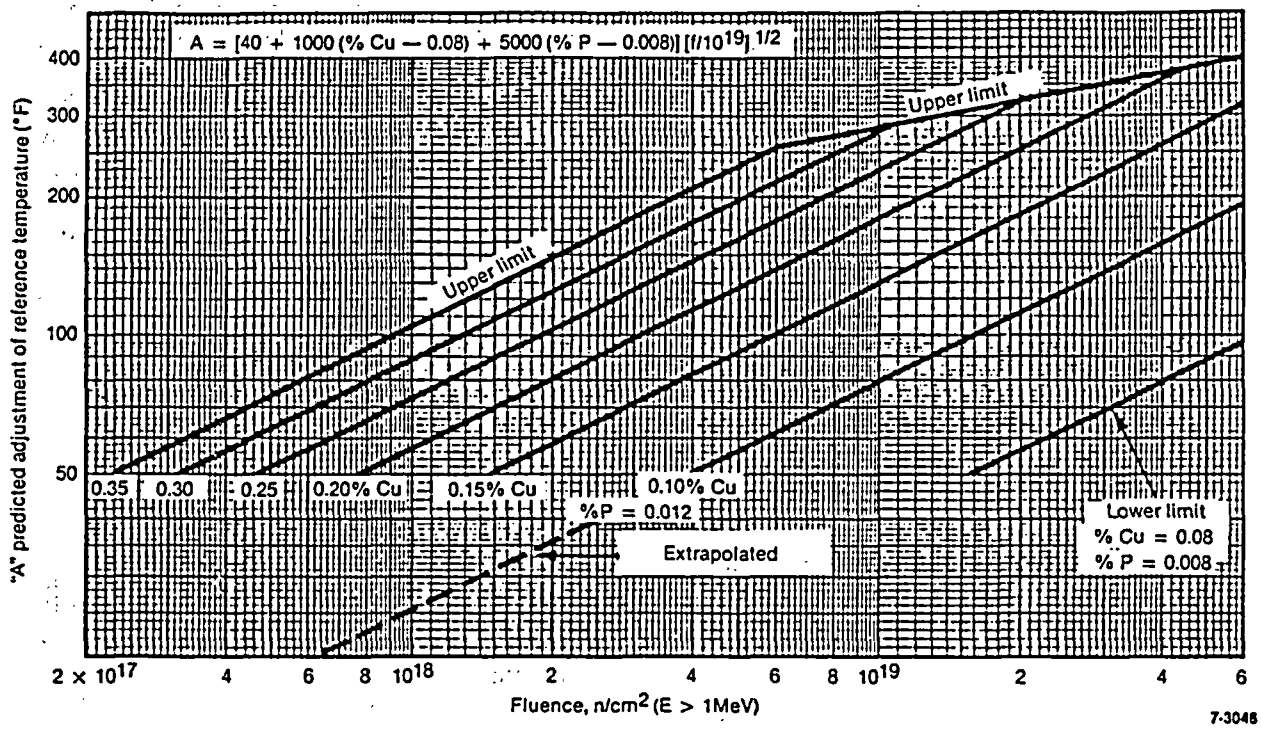


Figure 7.6. Predicted adjustment of reference temperature, "A", as a function of fluence and copper content. For copper and phosphorus contents other than those plotted, use the expression for "A" given on the figure.

$$\text{PLE margin} = \frac{\Delta(\phi t)}{\phi} = \frac{2 \times 10^{18} - 10^{18}}{10^{19}}$$

$$= 1 \times 10^{9} \text{ seconds}$$

Additional PLE margin = 32 years.

The above two examples show that certain RPV supports may have a PLE margin of several years. However, to develop a valid and more acceptable approach for calculating RPV support margins, more research is needed to determine the NDTT shifts due to both low energy neutrons (E <1 MeV) and lower irradiation temperatures [<450°F (232°C)]. The purpose of the above examples was to show the importance of the loading assumptions. However, it should be noted that the correlations for the temperature shifts in Regulatory Guide 1.99, Revision 1, are based on high energy neutrons (E >1 MeV), while the neutron spectra in the RPV supports have a very large component of low energy neutrons (E <1 MeV). Therefore, the NDTT shifts as developed from dpa gradients (similar to Equation 3, Regulatory Guide 1.99, Revision 2[15]) may be more appropriate then the use of Regulatory Guide 1.99, Revision 1.

7.3 Recommended Strategies for RPV Supports PLE

There are two basic options that can be simultaneously pursued for RPV supports PLE. They are administrative relief and monitoring and sampling.

The first option, administrative relief by way of the new GDC-4 will result in the elimination of design basis loadings because of a double-ended break of the reactor coolant piping. As previously mentioned, this one tactic alone provides a good means of extending the use of RPV supports.

The second option of monitoring and sampling allows an actual NDTT evaluation of the supports that have experienced operational neutron radiation exposures. Once the neutron environment of the RPV support has been assessed per the following ASTM Standard E 1035,[20] the corresponding shift in NDTT can be calculated,[5] based on the type of material employed in the support.

In ASTM Standard E 1035, it is recommended that all pressurized water reactors whose supports see a lifetime neutron fluence (E >1 MeV) that exceeds 1×10^{17} neutrons/cm^2 or 3×10^{-4} dpa consider monitoring as a first step in evaluating any possible NDTT in the supports.

The standard outlines state-of-the-art methods in dosimetry that are now possible to properly characterize the neutron field near the RPV supports. In particular, it describes how scrapings from the RPV or the supports themselves can be used as passive dosimeters through the use of a relatively new and very sensitive technique[21] known as helium accumulation fluence monitors (HAFM). In this technique, the helium atoms released from the reaction Cu(n,α) are counted and are a measure of the neutron field.

Furthermore, the standard suggests the refinements in the reactor physics calculations that are needed to properly assess the analysis of the integral dosimetry data and the final prediction of NDTT. For example, if the vessel supports do not lie within the core's active height, then an asymmetric quadrature set must be chosen for discrete-ordinates calculations that will accurately reproduce the neutron transport in the direction of the supports. Care also has to be exercised in constructing the quadrature set to ensure that ray streaming effects in the cavity air gap do not distort the calculation of the neutron transport. If the support system is so large or geometrically complex that it actually perturbs the general neutron field in the cavity, then the analytic method of choice is a coupled discrete ordinates/Monte Carlo calculation. The normal coupling for this type of problem is to perform the two-dimensional discrete ordinates analysis only within the vessel. The neutron currents generated by this analysis are then used to create the appropriate cumulative distribution functions in the final Monte Carlo analysis.

After the monitoring data have been unfolded, estimates of the exposure in dpa can be made using ASTM practices E 944 and E 1018. The dpa then can be converted to NDTT using the newest relations outlined by McElroy.[22] If these monitoring results indicate a potential for the PLE margin, then it also may be prudent to take actual samples of the RPV support to confirm the current NDTT by way of NDTT tests (cf. ASTM E 208). The structural considerations of removing suitable samples without any subsequent repair are discussed below.

Four basic RPV support types have potential for sampling if the reactor cavity is large enough to allow access. These include the column, bracket, cantilever, and neutron shield tank type (see Figures 7.1 through 7.4). Samples are assumed to be removed from an area adjacent to the mid-plane of the core usually considered to be the location of highest neutron fluence. However, it should be pointed out that this should be confirmed by way of reactor physics calculations and by way of the dosimetry monitoring outlined above or both. This is because new low-leakage fuel patterns may cause the maximum neutron fluence to occur off the reactor core's horizontal midplane. Based on the above assumptions, a preliminary review indicates that permanent removal of a specimen from the neutron shield tank type is not feasible because it is a fluid retaining vessel.

However, for the bracket, cantilever, or column type supports removal of a sample may be practical. The design basis for existing RPV supports are, in general, extremely conservative. This is, in part, because of the unrealistic loads or load combinations that may have been assumed in their design bases. Large design margins such as those previously explained regarding pipe break exist, as well as state-of-the-art refinements in analytical techniques used in the original design.

7.4 Other Factors Affecting PLE for RPV Supports

The above sections have discussed the primary area of concern with RPV supports for PLE, i.e., the potential catastrophic failure of embrittled supports. This section will address those other factors that, although they are not as significant as brittle failure, are still worth considering primarily because their synergistic effects in a nuclear power plant environment have not been the subject of detailed research programs.

- Corrosion. One of the factors present in all LWR containments is a greater than average humidity at an elevated temperature coupled with the presence of a radiation environment. The effect of the gamma radiation can be important in causing increased corrosion rates synergistically with the high humidity and temperatures. The long-term effects of such an environment on the integrity of the RPV supports should be examined because the total gamma dose alone on the supports will be over 5000 Megarads in the first 40 years of life. RPV support corrosion during the original license period is not expected to be a problem; only minor local pitting has been observed to date.

- Radiation damage to nonferritic parts of the RPV support system. Some support systems depend on a dry lubricant, e.g. Lubrite, which is located between the support and the RPV nozzle. This material has a radiation threshold dose of 2.2 x 10^3 Megarads. Therefore, any PLE plans should evaluate the actual radiation levels the lubricant has received and will receive over a PLE period. This evaluation would determine if supplement or replacement of the lubricant is needed.
- Stress corrosion cracking (SCC) of threaded parts in the sliding foot assembly. The threads are coated with Heresite. The threshold stress for initiation of SCC is 150 ksi while the applied tensile stresses are calculated to be about 20 ksi, therefore SCC of the sliding foot assemblies is not expected.
- Fatigue. The skirt type supports for BWR reactor pressure vessels are subjected to fatigue because the expansion and contraction cycles associated with the temperature- and pressure-induced expansion and contraction of the vessels during the plant startup/shutdown cycles. However, the fatigue usage factor during the first 40 years of operation is expected to be well below 1. (This degradation mechanism for the skirt-type supports is also discussed in Chapter 9 of this report.)

7.5 Summary, Conclusions, and Recommendations

This chapter of the report has discussed the primary controlling phenomenon for PLE for reactor pressure vessel supports. Five basic types of supports used in LWRs were examined: neutron shield tank, column, cantilever, bracket and skirt type supports.

The major conclusion reached was that the only important potential failure mode for the neutron shield tank, column, and cantilever type supports is catastrophic brittle failure due to irradiation embrittlement. Table 7.7 summarizes these findings and also points out other possible degradation mechanisms (corrosion), stressors, and in-service inspection methods. The bracket and skirt type supports will experience no significant NDTT shifts due to their location. Therefore, BWRs as a class of LWRs will have to address only fatigue for PLE because they are all skirt supported. Because some Babcock & Wilcox PWRs are also skirt supported, they, too, should be considered to be in this class.

A method to predict PLE margins for RPV supports based on a combination of the USNRC Regulatory Guide 1.99, various ASTM standards on dosimetry for RPVs, and Pellini's FAD was discussed in Section 7.2. In addition, it may be possible to get actual support material for NDTT tests. It was demonstrated that by using only the administrative relief for leak-before-break (GDC-4 of 10 CFR 50) that margins of several years could probably be realized.

As a result of the review conducted for this report, the following are recommendations for further study:

1. Develop fracture toughness and strength assessment data for the RPV support steels irradiated at temperatures less than 450°F (232°C). Develop a correlation of fracture toughness versus Charpy V-notch properties at temperatures less than 450°F (232°C) and as a function of dpa and neutron fluence (E >1 MeV).
2. Determine the range of radiation conditions (neutron spectra and flux levels) in and around shield tanks, and cantilever and column-type support structures.
3. Investigate the effects of the expected radiation levels due to extended operation on the lubricants between the RPV nozzles and supports.

Table 7.7. Summary of degradation processes for PWR and BWR RPV supports

Rank	Degradation Site[a]	Stressor	Degradation Mechanism	Potential Failure Modes	ISI Method[b]
1	Neutron shield tank at the core horizontal midplane elevation	Neutron irradiation, operating temperature, water chemistry	Neutron embrittlement, corrosion	Catastrophic brittle failure	Monitoring[c]
2	Column support at the core horizontal midplane elevation	Neutron irradiation, operating temperature	Neutron embrittlement	Catastrophic brittle failure	Monitoring and sampling
3	Cantilever support in the active height of the core	Neutron irradiation, tensile stresses, operating temperature	Neutron embrittlement	Catastrophic brittle failure	Monitoring and sampling
4	Threaded parts in sliding foot assembly	Tensile stresses, operating temperature	Stress corrosion cracking	Binding that causes possibly excessive stresses in the primary coolant system during heatup and cooldown	—
5	Dry lubricant in sliding foot assembly	Neutron irradiation, operating temperatures	Degradation caused by neutron irradiation	Binding that causes possibly excessive stresses in the primary coolant system during heatup and cooldown	—
6	Skirt support	Mechanical and thermal stresses	Fatigue	Ductile overload failure	—

a. The bracket and skirt RPV supports will experience no neutron embrittlement.

b. There are no national standard in-service inspection methods per se to determine state of degradation.

c. Monitoring of neutron field near RPV support.

7.6 References

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8. RANKING OF MAJOR BOILING WATER REACTOR COMPONENTS

V. N. Shah

This chapter identifies the major components of interest in commercial boiling water reactors (BWRs) for continuing safe operation and life extension. [Pressurized water reactors (PWRs) were discussed in Chapter 2.] In identifying these components, it is assumed again that the lifetime of the small, less-expensive components, i.e. sensors, controls, batteries, certain types of pumps, valves, motors etc., is much shorter than 40 years and these components will be maintained, refurbished, and replaced frequently. Therefore, the residual life of these small components is not significant to life extension of the plants and is not addressed in this section.

The major components reported here are identified and prioritized according to their relevance to plant safety only. The industry-sponsored Monticello pilot plant project has also identified components for plant life extension study.[1,2] However, the identification and prioritization criteria used by the industry pilot project were based on plant safety, reliability, and cost.

8.1 Key BWR Components

The two criteria employed to select the key BWR components are the same as those employed in Chapter 2 to select the key PWR components: to contain any release of fission products that may take place during any accident, and to maintain an acceptable level of radioactivity in the containment during normal operation. The pressure boundary components, containment, control rod drive mechanism (CRDM), reactor pedestal, and biological shield are included in the selected key components.

Major BWR Components

1. Containment and basemat
2. Reactor pressure vessel
3. Recirculation piping, safe ends, and safety system piping
4. Recirculation pump body
5. Control rod drive mechanisms
6. Cables and connectors
7. Emergency diesel generators
8. Reactor pressure vessel (RPV) internals
9. Reactor pedestal
10. Biological shield.

Many of the reasons for the above ranking of the major BWR components are the same as those discussed in Chapter 2 for the PWR components. However, the degradation of BWR pressure vessels caused by neutron embrittlement is less severe than that of PWR vessels, and therefore it is not ranked Number 1. Also, some of the degradation sites and the potential degradation mechanisms of the major BWR components are different than those identified for the PWR components and are discussed below.

The containment and basemat are ranked first because they are the most critical major components, as far as public safety is concerned. Most of the early BWRs have a Mark I pressure suppression containment system. This containment concept features a relatively small drywell connected by a system of vent pipes to another containment compartment commonly referred to as a wetwell. The other BWR containment concepts (Mark II and Mark III) also have a wetwell and drywell but are of larger size. The foundation of the drywell of the Mark I containment is made of concrete.[1] Two demonstration BWR plants, i.e., Big Rock Point (71-MWe capacity) and Dresden-1 (207-MWe capacity) have spherical metal containments without water suppression.

The potential degradation mechanisms for the Mark I drywell metal shells are corrosion of the outside surface and fatigue. The potentially life-limiting mechanisms for the suppression chamber are corrosion of inside and outside surfaces and fatigue of the torus metal shell and vent pipes.

The BWR reactor pressure vessel differs from the PWR vessel in the way it provides support to the reactor core. In the PWR vessel, the lower core support structure carries the weight of the reactor core and transmits it to the reactor vessel head flange. In the BWR vessel, the control rod drive (CRD) stub tubes provide vertical support for the CRD housings, guide tubes and mechanisms, and the fuel assemblies. Intergranular stress corrosion cracking (IGSCC) has been found at the weldments joining the stub tubes to the lower head. Also, the degradation of BWR reactor pressure vessels by neutron embrittlement is less severe than that of PWR vessels because there is a lot more water between the BWR fuel and vessel because of the location of the jet pumps. Most BWR vessels have an expected end-of-life fluence ($E > 1$ MeV) of about 5×10^{17} n/cm^2, while typical PWR vessels have an expected end of life fluence of 1×10^{19} n/cm^2. Therefore, vessel fluence and pressurized thermal shock are not significant issues in BWRs.[3] However, irradiation embrittlement cannot be totally discounted when assessing the residual life of BWR vessels. The new Regulatory Guide 1.99, Revision 2, takes into account Cu and Ni content in the vessel materials, and dictates higher transition temperature shifts than those previously determined by Regulatory Guide 1.99, Revision 1, for BWR vessels. Fatigue is the most likely degradation mechanisms for BWR reactor vessels.

The feedwater nozzles of the BWR reactor vessel constitute an important part of the primary coolant pressure boundary. Cracks have been found in the feedwater nozzles at Monticello and 17 other BWR plants.[4] The feedwater nozzles are subjected to relatively severe cyclic thermal loading because of turbulent mixing of incoming cold water [340 to 435°F (171 to 224°C)] and the hot water [545°F (285°C)] returning from the steam separators and dryers. The thermal loading is more severe during startup and shutdown because the incoming cold water is at a temperature of about 100°F (38°C). The main degradation mechanism is high-cycle thermal fatigue. The stainless steel cladding on the nozzle surface contributes to the fatigue cracking because of the different thermal expansion coefficients of the stainless steel cladding and the carbon steel base metal.

The key BWR piping components that are susceptible to aging are the recirculation piping, the safe ends, the feedwater piping, and the core spray spargers. After more than ten years of operation, several piping components in the Monticello plant have been repaired or replaced.[1] The core spray safe end was replaced in 1981, and some recirculation piping welds were weld overlayed in 1982. The recirculation piping and safe end were replaced in 1984. Similarly, the recirculation piping in the Pilgrim power station has been replaced.[5] Several other BWRs have experienced similar failures at the recirculation piping safe ends.[6] The potential degradation mechanisms for the recirculation piping and its safe ends are IGSCC, crevice corrosion, and thermal fatigue.[3] The feedwater spargers distribute the feedwater such that the flow is evenly distributed to the jet pumps. Sparger heads have experienced cracking at the flow holes because of fatigue and IGSCC.[4] BWR spray headers also have experienced similar fatigue and IGSCC problems. For example, the Peach Bottom spray header experienced significant cracking and has been repaired.[7]

The key locations of the potential degradation sites and potential degradation mechanisms for the recirculation pumps, CRDMs, cables and connectors, and emergency diesel generators are likely to be similar to those discussed for the corresponding PWR components. If there are any variations in the degradation processes, they will be reported later.

The key RPV internal components susceptible to aging degradation are the core shroud, jet pumps, fuel support pieces, steam separators, core plate, and steam dryers. The CRDM, feedwater piping and spargers, and core spray spargers that were discussed above also are key RPV internal components. The reactor internals provide positioning and support of control rods, fuel rods, and other components as in PWRs. However, the design of the BWR internals is quite different than that of the PWR internals. Their failure may cause binding of the control blades, thus prohibiting their insertion into the core. This binding may lead to an operational transient without scram. Failure of the BWR internals also may cause fuel rod cladding degradation, fuel relocation, wet steam because of failure of dryers, and loss of forced recirculation flow because of failure of the jet pumps. Type 304 stainless steel and Inconel 600 are the principal materials of the reactor internals that see high fluences. Neutron irradiation of Type 304 stainless steel will increase its yield and tensile strengths, decrease its uniform elongation, cause relaxation, lead to IGSCC, and reduce the Charpy impact upper shelf toughness. Neutron irradiation also reduces the low-cycle fatigue life of Type 304 stainless steel components. Among all the reactor internal components, the core shroud is most susceptible to thermal shock following a design-basis accident.[3] The main degradation locations in the jet pumps are the holddown beams, restrainer brackets, and sensing lines.[8] The holddown beam is fabricated from heat-treated Inconel 600 and has failed in several BWR plants because of IGSCC.[9] The restraining brackets have failed because of improper assembly, and sensing lines have failed because of vibrations. The jet pumps have been successfully repaired under field conditions, and therefore, the degradation of their subcomponents may not have a significant impact on the life extension of BWR reactor internals. The failure of the steam separators and steam dryers will result in poor quality steam passing through the steamlines to the turbines but probably will have no safety significance. The wet steam may cause erosion and corrosion in the steam piping and the first stage of the turbine blades.[10] Other potential degradation mechanisms for the BWR internal components are low-cycle fatigue, stress-corrosion cracking, and high-cycle fatigue.

The reactor pedestal and biological shield are made of concrete. The key locations and potential degradation mechanisms are the same as those for the concrete in the PWR containment structures. However, the reactor pedestal is subjected to high-cycle, low-amplitude vibration and thermal cycling. An additional loading on the biological shield is gamma-ray heating resulted from the proximity of this component to the core.

8.2 Summary, Conclusions, and Recommendations

The major BWR components were selected and ranked such that any release of fission products during an accident will be properly contained. The containment and basemat have been identified as the most important safety components in BWR plants. This means that a failure of the containment and basemat will have a major impact on the safety of the power plant. The other key components, according to their ranking, are reactor pressure vessel, recirculation piping and safe ends, recirculation pump body, CRDMs, cables and connectors, emergency diesel generator, RPV internals, and RPV supports and biological shield. The results presented in this chapter are based on past experience, as reported in the available literature. These results will be updated as more data are made available.

Table 8.1 summarizes the aging related information on the key BWR components discussed in this chapter. It lists the components and gives reasons for their ranking. Table 8.1 identifies the most likely degradation site and other degradation sites for each component and also lists the most likely degradation mechanism and other potential degradation mechanisms.

The major components reported in this chapter are selected and prioritized according to their relevance to plant safety only. These criteria are different than the ones used in the industry pilot study. Therefore, the ranking of the major components presented in this chapter is somewhat different than the one in the pilot study. However, all the major components identified in these chapters are also ranked high in the pilot study. Detailed discussions of the degradation processes for BWR pressure vessels and recirculation piping are presented in Chapters 9 and 10, respectively. Discussion of the degradation processes of both PWR and BWR RPV supports was presented in Chapter 7.

Table 8.1. Key BWR components for residual life assessment

Rank	Component	Reasons for Ranking	Degradation Sites (most likely, others)	Degradation Mechanisms (most likely, others)
1	Containment and basemat	Public protection during an accident	Concrete, metal liner, and reinforcing steel in suppression pool and primary containment wall, base metal, welds and vent pipes in metal containment	Environmental degradation of concrete, fatigue and corrosion of metal containments
2	Reactor pressure vessel	Primary pressure boundary, poor operating experience with RPV nozzles	Nozzles, closure studs, beltline region, welds at the stub tubes	Thermal fatigue, neutron embrittlement, IGSCC
3	Recirculation piping, safe ends, and safety system piping	Primary pressure boundary, relatively poor operating experience	Safe ends, austenitic stainless steel fittings	IGSCC, thermal aging
4	Recirculation pump body	Primary protection from any internal failure of pump elements, primary coolant pressure boundary	Heat-affected zones near weldments in the wall	Thermal aging, corrosion fatigue, crevice corrosion
5	CRDM	Failure may lead to a reactivity-initiated accident, an anticipated transient without scram, or a small-break loss-of-coolant accident	Drive rod assembly, control rod pressure vessel	Wear and thermal embrittlement
6	Safety-related cables and connectors	Active during normal operation in mitigating operational transients and accidents	Cable insulation, inserts in connectors	Thermal aging of insulation, thermal embrittlement and corrosion ofconnectors
7	Emergency diesel generator	Needed to operate critical safety equipment in the event of a loss-of-offsite power	Governor in the instrumentation and control system, cooling and lubrication system pumps and piping, fuel injector pumps, turbocharger, generator windings	Fatigue and vibrations
8	Reactor internals	Failure may cause fuel failure or problems in scraming the reactor	Core shroud, top guide plate, core plate, holddown beam in a jet pump, feedwater and core spargers	Irradiation-assisted stress corrosion cracking, intergranular stress corrosion cracking, fatigue
9	Reactor pedestal	Failure will challenge integrity of RPV	Concrete and reinforcing steel	Thermal cycling of concrete, corrosion and fatigue of reinforcing steel
10	Biological shield	Maintains acceptable level of radioactivity in containment	Inside surface of the biological shield at the core horizontal midplane level	Neutron irradiation, gamma-ray heating, environmental degradation of concrete, corrosion of reinforcing steels

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9. BOILING WATER REACTOR PRESSURE VESSELS

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9.1 Description

This discussion for boiling water reactor (BWR) pressure vessels is not as detailed as that for the pressurized water reactor (PWR) pressure vessel presented in Chapter 3. However, the same important degradation mechanisms (namely fatigue and irradiation damage) are considered for the BWR vessels, as will be discussed. Primarily, the recollections and experience of one of the authors are presented; therefore, some of the details and specifics should be considered as close approximations.

BWR vessels have been manufactured throughout the years with the same basic geometry shown in Figure 9.1. They consist of a cylindrical shell and a hemispherical bottom head containing both the control rod drive (CRD) penetrations and the in-core instrumentation penetrations. The top head also is hemispherical and includes nozzles with bolting flanges attached. One of the nozzles is for venting (pressure relief) and top head cooling spray, and the others are spares. The top head is bolted to the cylindrical shell by means of ∿6-in. (150-mm) diameter studs. The flanges are butt welded to the shell and to the head and are furnished in one piece. The earliest BWRs were of several sizes/geometries and represented demonstration and small power reactors (i.e., Dresden-1, Big Rock Point, Humboldt Bay, and La Crosse). The early commercial-generation BWRs consisted of three sizes (not including Duane Arnold which was smaller). These sizes were ∿201-in. (5.11-m), 218-in. (5.54-m), and 251-in. (6.38-m) diameters. For example, the Vermont Yankee and Monticello vessels were 201-in. (5.11-m) diameter, Zimmer and Brunswick-1 and -2 were 218-in. (5.54-m) diameter, and Quad Cities-1 and -2, Susquehanna-1 and -2, and Limerick-1 and -2 were all 251-in. (6.38-m) diameter. These are just examples and do not include all of the plants of any given size.

The latest BWR model, which was designated as BWR/6, was built in two sizes. These were 218- (5.54-m) and 238-in. (6.05-m) diameters. Examples of plants of these sizes are River Bend-1 and Clinton-1 for the 218-in. (5.54-m) diameter and Grand Gulf-1 and Perry-1 for the 238-in. (6.05-m) diameter. Quite a few of the larger diameter vessels were constructed for utilities in this country (many of which have been cancelled), while the other 218-in (5.54-m) diameter vessels were built for overseas plants (i.e., Laguna Verde and Kuo Sheng).

The predominant vendors of BWR vessels in this country have been Babcock & Wilcox (B&W), Combustion Engineering (CE), and Chicago Bridge and Iron (CB&I). One foreign-produced vessel from Babcock/Hitachi was fabricated for Hope Creek. One of the vessels finished by CB&I was partially fabricated by Rotterdam Dockyard. Rotterdam Dockyard did have a contract to construct one of the vessels for the Black Fox units that were cancelled. In total, CB&I built most of the BWR vessels used in the United States.

All of the major nozzles of the vessel where placed in the cylindrical shell, but they were not located at the elevation of the reactor core (see Figure 9.1). The nozzles in the lower portion of the shell were, in general, limited to the primary coolant recirculation inlet and outlet nozzles which were located in a circumferential band a short distance above the attachment of the cylindrical shell to the bottom head. Nozzles above the core region were designed for core spray, feedwater, and the steam outlet. The largest nozzles were the two for the recirculation outlet lines (primary coolant outlet) at the bottom part of the shell; they were typically 26 in. (660 mm) in diameter. Few, if any, of the other nozzles exceeded 12- to 15-in. (305- to 380-mm) diameters.

All the nozzles in the vessels built by CB&I were forged and provided a lip at the shell end of the forging to permit butt welding of the nozzle to the shell. Reactor vessels furnished by at least one other fabricator included nozzles which were forged but did not contain a lip so that the attachment weld of the nozzles to the shell was a corner weld.

The earliest BWR reactor vessel was Dresden-1. The plates used in its construction were procured to the requirements of American Society for Testing and Materials (ASTM) A300 for fracture toughness (Charpy V-notch) testing and American Society of Mechanical Engineers (ASME) Code Case 1280 which provided requirements for fine grain melting. Before the introduction of the SA533 Grade B, Class 1 (SA533B-1) plate specification, SA302 Grade B (SA302B) plate was used. All vessels since about 1965 have been constructed of SA533B-1 material (this includes the plates used in the shell and in both heads). It appears that all BWR vessels have been constructed using plates for the shell and the heads. That is, there have been no forged rings, nor forged head segments, used in this country.

The forging materials used for the nozzles in Dresden-1 were A105 II (equivalent to present A105).

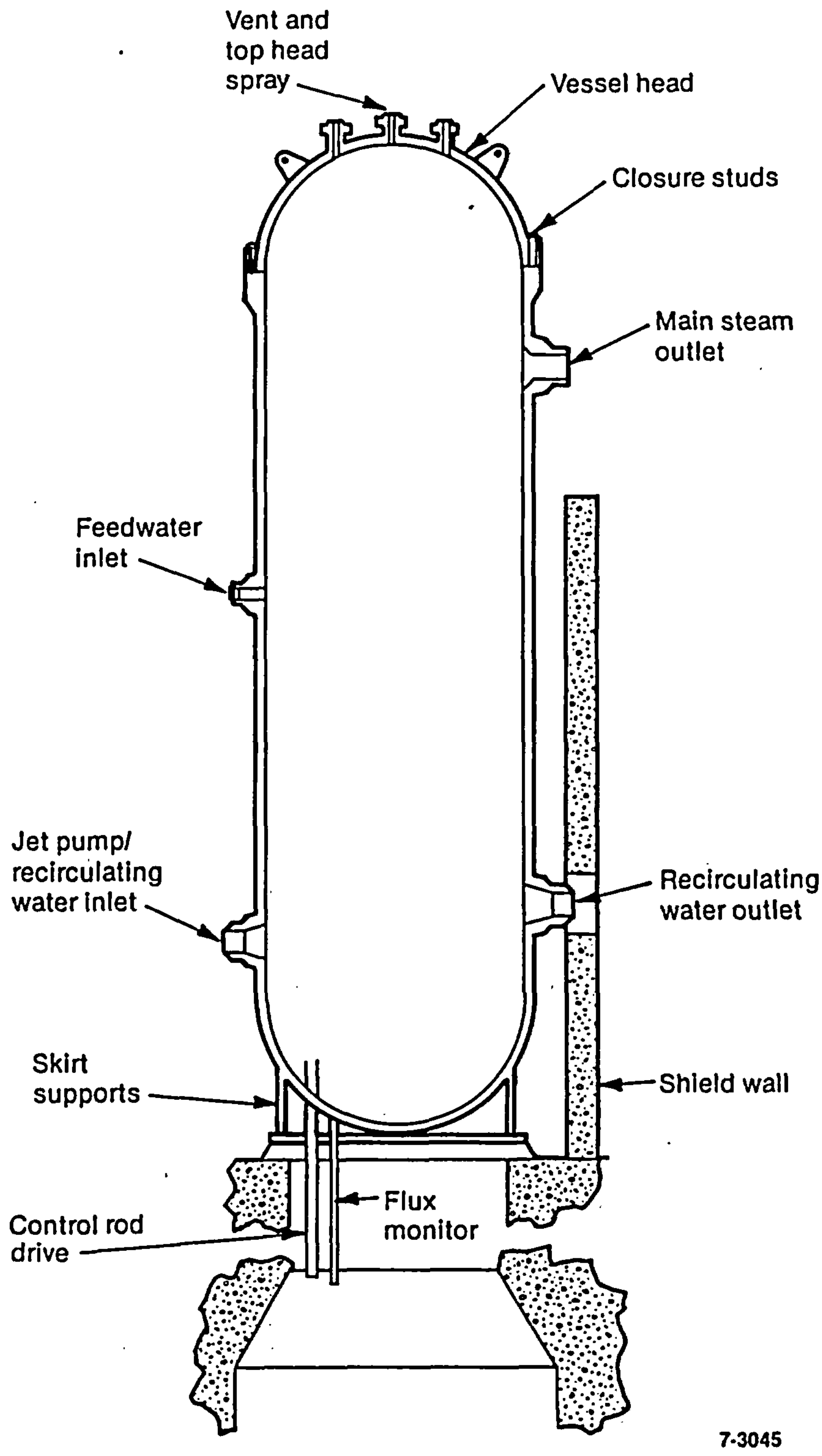

Figure 9.1. Typical BWR pressure vessel.

Probably, this material was used for most of the SA302B vessels, before the introduction of the SA508 specifications. All of the more recent vessels have used SA508, predominantly Class 2, (SA508-2) although some Class 3 material (SA508-3) has been used.

Before the introduction of the SA540 bolting specification, the studs were furnished to an A193-B7 specification. Subsequently, the studs have been SA540, B22, or B23 for the 130 ksi (900 MPa) yield strength level. It is possible that some studs were furnished at the 140 ksi (965 MPa) yield strength level.

Until about 1974, the CRDs were attached to a stub tube that was welded to the inside of the bottom head of the reactor vessel. Until about 1966 or 1967, the stub tubes were made of stainless steel. Thereafter, they were made of Inconel (Alloy 600). From about 1974 to the present, the CRDs were welded directly to the bottom head without an intervening stub tube. The in-core instrumentation penetrations were always welded directly to the bottom head. The vessel is supported by a skirt welded to the bottom head just below the attachment of that head to the cylindrical shell, as is shown in Figure 9.1. The design of that skirt in terms of load-carrying capability is governed by the weight of the vessel and its contents and includes consideration of the operating basis earthquake loadings.

As mentioned earlier, all fabricators made the heads and cylindrical shells of BWRs with formed and welded plate. CB&I purchased the heat-treated plate from the mill, and then cold formed and welded it together. The welding used proprietary welding filler metals that generally matched the chemistry and properties of the base material. The other primary fabricators of BWR vessels in this country, B&W and CE, purchased unheat-treated materials from the plate mills, hot formed the material, electroslag welded the joints, and heat treated the complete assembly after the rings were complete. This heat treatment consisted of austenitizing followed by tempering. All vessel manufacturers welded the circumferential seams by the submerged arc process and completely stress relieved the vessel in accordance with ASME code requirements after the welds were made. In some cases, an intermediate postweld heat treatment may have been made for individual ring assemblies before joining them in a complete vessel.

The earliest vessels were designed to the requirements of the ASME Code, Section I, and the 1270 series of code cases. All vessels manufactured after 1963 were furnished to the requirements of Section III of the ASME Code. The mechanics of making the design and satisfying the code analytical requirements are pretty much uniform and conventional regardless of the vendor. Of course, the analytical methods did change as available computer programs became more developed. An example of this evolution would be the evaluation of the discontinuity analysis from the old beam on an elastic foundation analysis to computer programs for closed form numerical integration of the shell equations and finite element methods.

Problems that have occurred throughout the years have resulted in changes to both design and materials. For instance, the safe ends on the nozzles of BWRs were originally forged of austenitic stainless steel and stress relieved after welding to the vessel. The occurrence of intergranular stress corrosion cracking (IGSCC) in the primary coolant lines of BWR plants led to the removal of these postweld heat-treated austenitic safe ends and replacement with safe ends made of austenitic materials that had not been stress relieved. Additionally, some safe ends were fabricated of Inconel. Similarly, all of the nozzles of BWRs were originally stainless steel overlayed. However, thermal fatigue cracking was experienced at Millstone-1 as a result of bypass flow and a loose thermal sleeve in the feedwater line. Subsequently, the BWR vessel nozzles have not been overlayed.

There also has been an evolution in the design of the attachment to the inside of the nozzles for thermal sleeves and the recirculation inlet lines. As mentioned above, there was cracking at Millstone-1 because of bypass of feedwater flow around the safe end of that nozzle. The thermal sleeve was not integrally attached to the nozzle. All of the lines in these vessels have since been repaired to provide for a welded attachment to preclude this problem. Similarly, the attachment to the inside of the recirculation inlet nozzle provides flow from the inlet line to the jet pump risers. Initially, these jet pump riser lines were close-fitted to the inside of the nozzle and welded at the end (a detail that was a combination of a groove and a fillet weld). However, stress corrosion cracking of the safe end (that is actually where these lines are welded) at the Duane Arnold plant created a concern about IGSCC of Inconel and crevice conditions. Therefore, most of the safe ends have been replaced, eliminating this crevice condition.

9.2 Stressors

The basic design requirements for the BWR vessel were 1250 psi (8.62 MPa) pressure and ∿550°F (288°C) temperature. The actual operating conditions at steady state are around 1,000 psi (6.89 MPa) and 540°F (282°C). The design specifications included the transients necessary to heatup and cooldown the vessel from normal operating conditions and a variety of normal and abnormal conditions that might be expected throughout the life of the vessel. The magnitude of the changes were provided as were the rate at which they occurred. Pressure loading in excess of 1250 psi (8.62 MPa) design level occurs during safety valve discharge when the pressure is permitted to rise to 1375 psi (9.48 MPa).

The skirt support for BWR vessels is shown schematically in Figure 9.1. One of the key design considerations here is that of earthquake loadings. Additionally, the attachment of this skirt to the bottom head requires a degree of refinement beyond that normally encountered in pressure vessel construction because of the necessity for meeting the code stress limits for the primary plus secondary stresses and fatigue stresses that occur as a result of the pressure/temperature cycles at that intersection.

All of the nozzles were designed to carry specified pipe loads and the bimetallic or trimetallic safe end attachment welds to the nozzle forging area were analyzed for differential thermal stresses, as well.

The core support structure in a BWR consists of an annular plate welded at its outer periphery to the bottom of the cylindrical shell immediately above the shell to the bottom head attachment weld. The inner periphery of this annular plate is supported by a series of heavy Inconel posts or columns. The attachment of the outside periphery of the annular plate and the bottom of the Inconel columns (at the attachment to the bottom head) is analyzed in detail for all mechanical, thermal, and pressure conditions. Likewise, the attachments at the upper portion of the cylindrical shell for the core spray spargers and the feedwater spargers are thoroughly analyzed for specified mechanical and thermal loads.

9.3 Degradation Sites

As mentioned earlier, three of the highest ranking potential degradation sites have been eliminated in the newer plants by design changes and repaired in most of the older plants. These include: austenitic stainless steel safe ends that have been sensitized by postweld heat treatment, crevice areas where thermal sleeves are attached to Inconel or stainless steel safe ends, and austenitic cladding on the inside corners and surrounding regions of the nozzles (that experience fairly wide temperature swings). Another possible area of concern that has been eliminated in later design modifications is the CRD penetration stub tube to vessel weld heat-affected zone; earlier plants that used stainless steel stub tubes were susceptible to stress corrosion cracking at this weld (leading to through-wall leakage). Failures did occur at two plants. The most recent BWR-6 reactor pressure vessels do not have stub tubes and the guide tube is welded directly to the bottom head. This decreases the number of potential degradation sites.

Regardless of whether the above design changes have been made, the ranking of potential degradation sites would still vary from plant to plant, depending on how they are operated. For example, one of the more critical items would be the closure head studs at the main closure flange. Because the greatest range of stress amplitude in the studs is produced during installation and removal of the head (and the use of stud tensioners associated with those operations), the number of times these occur in a specific plant would have a great effect on where the studs fall in the ranking of degradation sites. Thus, real operating history rather than design numbers and magnitudes of transients is necessary in assessing critical degradation sites.

The threaded stud holes in the flanges without bushings are subjected to wear because the flange material is considerably softer than the studs. In flanges with bushings, the threads for the closure studs and flange bushings could be subjected to fretting. However, the studs are removed from the flange infrequently. In addition, the use of stud tensioners for tightening makes fretting during this operation unlikely.

The design of the flange assembly for most, if not all, BWRs includes provision for sufficient preload to prevent slippage between the flange surfaces during operation. Therefore, the flange surfaces are unlikely to suffer physical damage except from mishandling during removal and placement of the top head. The O-rings are placed in the O-ring grove between the flange surfaces to provide sealing. The O-rings have a soft metal layer on the outside to enhance sealing and reduce wear, and are intended to be replaced when the head is removed.

Another important potential degradation site is the outer end of the nozzle forgings, and the attached safe end, for those nozzles that have water with large temperature variations flowing through them. The most severe nozzle of this type is the feedwater nozzle. The specified flow temperatures for this nozzle go from essentially the freezing point of water to the operating temperature of the vessel. The best operating example of problems caused by flow through this nozzle is Millstone-1 where nozzle corner cracking was found. The bypass of the feedwater around the thermal sleeve at this nozzle resulted in a temperature swing of ~200°F (110°C) at approximately 1-Hz frequency developing high-cycle fatigue cracking. Of course, the upper end of that temperature swing was when there was no bypass flow; therefore, this temperature range was larger than the variation in the line temperature itself.

In a general sense, all locations of dissimilar metal welds will be most sensitive to thermal cycling. The effect of this thermal cycling will be compounded by any mechanical loads occurring at the same location. Therefore, the most important areas in this regard would be the nozzle safe end attachments and the attachment welds to the pressure boundary for the core support structure.

The welds and the base metal in the reactor beltline region are the main sites subjected to radiation embrittlement. It should be noted that repair welds formed during construction and fabrication are also present sometimes in the beltline region of BWR vessels.

External attachments such as the support skirt and the refueling bellows skirt are subjected to cyclic loading caused by differential radial thermal-induced expansion during startup/shutdown. However, neither of these locations are fatigue limited, i.e. fatigue usage factors for these locations are far smaller than 1.0. Rather, the limiting condition is primary plus secondary stresses being controlled by the 3-S_m limit. There are some corner weld joints in the refueling bellows skirt assembly that may be fatigue limited for some vendor's designs.

9.4 Degradation Mechanisms

As indicated in the previous discussion, the primary degradation mechanism will be fatigue (induced through a combination of thermal and mechanical cycling). This cycling also has an effect on the main closure studs and is compounded by the additional cycles of tensioning and untensioning during top head removal. The threads for closure studs, flange bushing, and stud holes in flanges without bushing are subjected to wear, fretting, and corrosion, as discussed earlier. External attachments such as the support skirt and the refueling bellows are subjected to cyclic loading, as discussed in the preceding section.

There should not be any locations in a BWR vessel except the earlier ones having stainless steel stub tubes, where IGSCC should be a problem, because most (if not all) of these problem areas have been previously identified and modified. However, continued monitoring of these same areas should be maintained to ensure against any reoccurrence. Water chemistry control used to mitigate the potential for IGSCC in the recirculation piping also would reduce any likelihood for problems in the reactor pressure vessel.

All of the welds in the ferritic steel components have been postweld heat treated. Therefore, residual stress effects should not contribute to degradation. However, the welds joining Inconel, stainless steel, and the two materials to each other have not been stress relieved. An exception to this last statement would be the build up of these materials that is made for the attachment of Inconel (or stainless steel) to the ferritic material. These locations would be at the outboard end of the nozzle forgings and where the core support structure is attached. However, even in the locations where some residual stresses may exist, there is no need to be concerned with excessive fatigue damage.

Radiation embrittlement is not as severe a problem with BWR vessels as it is with PWR vessels (see Chapter 3). The fairly large diameters of the BWR vessels, and the amount of water shielding the shell from the reactor core, help alleviate this problem for BWRs. However, radiation damage can eventually become a limiting factor, depending on material chemistries and actual accumulated fluence. Most BWR vessels have an end-of-life fluence ($E>1$ MeV) of about 5×10^{17} n/cm^2, although some of the older vessels can be a bit higher; typical PWR vessels have end-of-life fluences above 1×10^{19} n/cm^2. However, irradiation embrittlement should not be totally discounted when assessing life extension of BWR vessels. From an operational standpoint, the effects of radiation damage can be more of a nuisance. For example, with the new Regulatory Guide 1.99, Revision 2 (discussed in detail in Chapter 3), hydrotesting of BWR vessels after experiencing moderate radiation exposure becomes more difficult because of the higher temperatures required for the hydrotests. However, the data used to develop the Regulatory Guide 1.99, Revision 2 strongly dictate higher transition temperature shifts than those previously used for BWR vessels at these low fluence levels.

9.5 Potential Failure Modes

Because pressurized thermal shock and low-energy ductile tearing are extremely unlikely in a BWR vessel because of the low neutron irradiation conditions in the beltline region, the only final failure mode is plastic overload leading to a leak in a region where a crack exists (or failure of a stud). Thus, the conditions leading to this final failure mode are fatigue or corrosion induced. All parts of the reactor vessel for which cyclic loading has been specified have taken fatigue into account and have fairly low usage factors (almost always below 0.6 over the intended 40-year life). However, as indicated in Chapters 3 and 5, the calculation of fatigue usage factors can vary and exact comparison of numbers can be misleading. Consideration of abnormal loading conditions that currently would be considered as Levels C or D (emergency or faulted) did not control any of the design features of a BWR vessel. In the earlier BWR vessels with stainless steel stub tubes, IGSCC-induced failures have resulted in leakage through the CRD penetrations.

9.6 In-service Inspection and Surveillance Methods

The PWR vessel in-service inspection (ISI) methods and requirements of Section XI of the ASME Code were presented in Chapter 3 and are the same for BWR vessels. However, many older BWRs have very limited accessibility for external ISI of the vessel. Typically, 75 to 90% of the vessel weld lengths are exempted because of inaccessibility. The only alternative is ISI methods of examination from the inside surface. This is of particular importance at the beltline welds. There were minor and major repairs to shell plates during construction but some of these cannot be inspected because of limited accessibility. This is also true for some of the pipe-to-nozzle welds that were not configured to facilitate ISI. This was changed in later reactors after the requirements of ASME Section XI were published.

Surveillance for irradiation embrittlement is dictated by federal law for BWRs as well as PWRs. Therefore, monitoring of actual changes in Charpy V-notch and tensile properties with regard to accumulated fluence for the most critical vessel materials is underway.

9.7 Summary and Conclusions

A summary of the degradation sites and mechanisms, stressors, potential failure modes, and ISI methods for BWR vessels is presented in Table 9.1. Most of the extensive conclusions listed in Chapter 3 also are applicable here. The main difference is the much less important effect of irradiation embrittlement for BWR vessels. However, this damage mechanism is present and should never be ignored.

Other conclusions worth restating here are: nozzle fatigue usage should be monitored more closely with regard to refined cycle counting and transient severity so that better estimates of actual fatigue usage factors can be determined, the closure studs should be examined closely for replacement near the normal 40-year end of life, and the large-nozzle, CRD, and other small-nozzle penetration welds should be carefully inspected.

Table 9.1. Summary of degradation processes for BWR reactor pressure vessel

Rank of Degradation Site	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI/Surveillance Methods
1	Nozzles (including instrument and CRD penetrations) plus safe end welds	Mechanical and thermal stresses	Fatigue crack initiation and propagation, IGSCC	Ductile overload leading to a leak	All large nozzle welds inspected volumetrically at each interval; visual, external surface inspections of small nozzles/penetrations
2	Closure studs, flange bushings, stud holes	Mechanical and thermal stresses	Fatigue crack initiation and propagation, fretting, corrosion	Ductile overload failure (can be replaced)	Volumetric and surface inspections of all studs, threads in flange stud holes and bushings
3	Beltline region	Irradiation embrittlement	Neutron irradiation (extent depends on vessel materials)	Ductile high-energy tearing leading to a leak (not a serious problem)	100% volumetric inspection; surveillance as required by federal law
4	External attachments	Mechanical and thermal stresses	Fatigue	Ductile overload failure	Volumetric and surface inspections

10. BOILING WATER REACTOR RECIRCULATION PIPING

H. Mantle

There has been an increasing incidence of intergranular stress corrosion cracking (IGSCC) failures in boiling water reactor (BWR) recirculation piping systems in the past 10 years: first in highly stressed small-diameter piping and later in large-diameter piping (12 in. and greater). While an actual pipe severance has never occurred, the problem has impacted plant availability and resulted in a substantial increase in radiation exposure to maintenance and inspection staff. Although IGSCC has been the Achilles' heel of BWR plants, great strides have been made in understanding the conditions under which it occurs and in developing both short- and long-term solutions. Other concerns that may affect life extension have been raised (such as low-temperature aging of duplex stainless steel castings, crevice corrosion, and thermal fatigue), but these have not yet been shown to be significant problems in BWR recirculation piping systems.

10.1 Description

The recirculation system draws coolant from the bottom of the reactor vessel through an outlet nozzle and downcomer spool into the recirculation pump. The pump discharges this coolant into a ring manifold that feeds the coolant by way of multiple risers to the associated inlet nozzles and jet pumps, thus completing the loop. The reactor recirculation system consists of two independent loops. Primary piping in each recirculation loop in a large BWR reactor ($\simeq$1100 MWe) consists of a 28-in. pump suction line, a 28-in. pump discharge line, a 22-in. pipe riser manifold, and five 12-in. taps (ten total taps, five for each loop) to the ten jet pumps inside the reactor. A simple schematic of the basic arrangement is shown in Figure 10.1. Key components include safe end nozzle welds, suction and discharge valves, the recirculation pump, and the ring manifold.

Historically, BWR recirculation piping systems have been fabricated from wrought, AISI types 304 and 316 (piping) and cast types, CF8 and CF8M, (pumps and valves) stainless steels. Replacement piping has been made from Type 316 NG (nuclear grade) that has been shown to be much more resistant to IGSCC than the conventional grades.

10.2 Stressors

Stressors include welding heat (sufficient to produce a sensitized microstructure), tensile stresses (both residual and applied), oxygenated cooling water, temperature and cyclic thermal stresses. These factors, either in combination or separately, can produce IGSCC, thermal fatigue, thermal embrittlement, and crevice corrosion.

10.3 Degradation Sites

Degradation sites include weld heat-affected zones and furnace sensitized components (safe ends), weld metal, duplex stainless steel castings (elbows, valves, T-joints), areas in the system with high thermally induced stresses predicted by stress rule index analysis[1] (nozzles, safe ends), and crevices at safe ends shielded from the bulk fluid environment. Additionally, based on the interpretation of the North American Electric Reliability Council's 1984 Component Cause Code Report,[2] the recirculation pump also is a primary degradation site (see Table 10.1). The report data indicate that the recirculation pump and driver have a greater outage frequency than the reactor coolant system valves and piping. The cause for greater outage frequency is not reported in Reference 2 and is probably not related to a loss in pressure boundary integrity.

10.4 Degradation Mechanisms

As discussed in previous surveys of aged nuclear power plant facilities,[3] degradation mechanisms most responsible for current aging-related failures can broadly be categorized as erosion, corrosion, vibration (fatigue), and foreign material related. Within the recirculation piping system of BWRs, failures (leaks) have primarily been related to IGSCC. Of secondary concern are crevice corrosion, low-temperature aging (embrittlement) of duplex stainless steel, and thermal fatigue.

Degradation resulting from IGSCC has essentially been controlled by way of a multidirectional effort, because IGSCC of austenitic stainless steels requires the combined existence of the following three conditions:

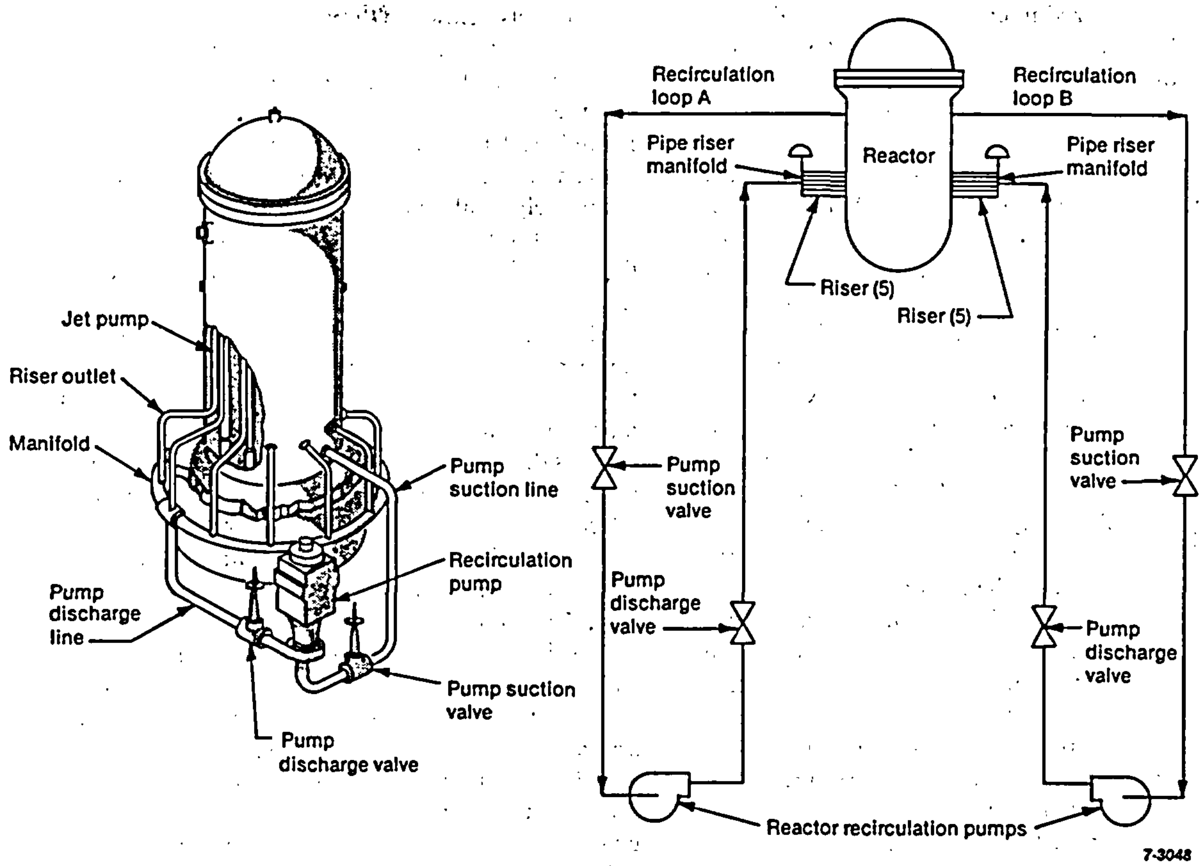

Figure 10.1. Schematic of a BWR recirculation piping system.

Table 10.1. Outage frequency: valves and piping versus pumps and drivers (years 1975-1984, 29 units, 230 unit years)

	Reactor Coolant System Valves and Piping	Reactor Coolant Recirculation Pump and Drivers
Forced	0.44	0.56
Unplanned derating	0.26	1.89
Forced maintenance, planned and equipment derating	0.93	2.79

1. A sensitized microstructure
2. A local tensile stress that exceeds the bulk alloy's yield stress
3. A corrosive (oxygenated) environment.

Even though these conditions theoretically exist in the heat-affected zones of most BWR recirculation piping welds, only a very small percentage of these welds have cracked. The primary factor that determines whether a particular site will develop IGSCC can therefore be inferred to be the existing local tensile stress since it is the most variable factor. It is also the most difficult to quantify, because it is made up of both applied (design loads) and residual (weld shrinkage, forming, machining, and grinding) stresses. Although IGSCC can be eliminated by controlling any one of the above three factors,[4] the current practice is to attack two or (in some cases) all three of the factors.

The first factor, sensitization of the base metal and weld heat-affected zone, is affected by the chemical composition of the base material (primarily carbon content) and the total thermal history (high weld heat inputs increase sensitization). The development of nuclear grade stainless steels with maximum 0.02 wt% carbon, nitrogen additions, and tightly controlled levels of impurities has produced a material much less susceptible to sensitization and thus more resistant to IGSCC. Installation contractors and nuclear steam system suppliers also have attempted to control sensitization by rapidly cooling the pipe interior after deposition of the root pass. This technique is called heat sink welding. This treatment reduces the degree of weld-zone sensitization and produces a more favorable pattern of residual stress on the pipe inside surface.

The second factor, tensile stress, can be mitigated by way of induction heat stress improvement (IHSI) and last-pass, heat sink welding. The first of these two techniques must be closely controlled to achieve the desired effect (residual compressive stress at the inside weld surface), particularly with welds made by the gas tungsten arc process.[5] It also has been suggested that IHSI may sensitize pump and valve body castings adjacent to the weld zone.

The third factor, oxygenated cooling water, can be attacked by careful control of water chemistry. Under steady-state conditions, BWR coolant contains 200 to 400 ppb dissolved oxygen. This level can rise to ∿8000 ppb under start-up conditions, thus further increasing the susceptibility to IGSCC. Because oxygen is constantly being created from the water coolant by way of radiolysis in the reactor core, the only practical method to control the oxygen to levels below the threshold for IGSCC within the recirculation loop is to inject hydrogen into the cooling water. This produces an environment favoring the recombination of hydrogen and oxygen. Ljungberg[6] has shown that levels of 3 to 5 ppb oxygen are achievable with this method. IGSCC is completely inhibited at levels below 5 ppb oxygen.

Also of concern for life extension of BWR systems is the effect that low-temperature aging has on duplex (austenitic-ferritic) stainless steel castings (and possibly weld metal) at operating temperatures and the potential for thermal embrittlement in localized areas. Exposure to temperatures in the 560 to 700°F (293 to 371°C) range promotes the formation of secondary phases (such as the G-phase, Type X, and alpha-prime phase) and carbide growth in these materials. The embrittling effect can be observed on materials with >10% ferrite. Materials with >20% ferrite show drastic reductions in impact energy when aged at higher temperatures [i.e., 750 to 850°F (399 to 455°C)]. Low-carbon grades in general show greater resistance to this embrittling effect, probably because of a reduction in the formation of chromium carbides and other secondary carbide phases. There also is some evidence to suggest that careful initial heat treatment of cast duplex stainless steel may produce a casting with superior unaged impact energies[7] such that even after aging significant fracture toughness (measured by room-temperature, Charpy V-notch energies) remains. Typically, these duplex stainless steel components are extensively weld repaired with attendant ferrite dilution in the weld zone. Detailed records of welded repairs and detailed ferrite surveys are not available. Ferrite measurements and monitoring should be done on the inside surfaces, which requires disassembly and extensive isolation provisions or draining the vessel after unloading the core. Further study of the thermal embrittling of the duplex stainless steel castings and particularly the weld material is recommended.[8] Additional information on the thermal embrittling of the duplex stainless steel components is discussed in Section 5.3.2, Chapter 5.

Crevice corrosion has been mentioned as a potential degradation mechanism that could occur over time at locations such as shaft sleeves and the underside of socket welds if a corrosive environment and a difference in oxygen concentration existed. To date, crevice corrosion has not been reported as a problem in BWR recirculation piping. This is probably a result of the tight controls maintained on water conductivity and oxygen levels in this system.

Thermal fatigue and corrosion-assisted fatigue also have been mentioned as potential degradation mechanisms. Frequent reactor scrams and unscheduled outages produce transient thermal stresses on both the cooldown and heatup cycles. Highly stressed hot spots need to be defined and a data base established to confirm whether this effect has significance for BWR recirculation piping. The weld overlays probably deserve specific attention. Fatigue considerations are currently being studied for carbon steel systems, but stainless steel systems (such as the BWR recirculation system) have not yet been addressed.[9]

10.5 Failure Modes

Failure modes produced by the failure mechanisms above would be primarily through wall defects resulting from cracks (IGSCC, embrittlement, or fatigue) or material wastage (crevice corrosion). So far, cracks in recirculation piping materials have satisfied a leak-before-break criteria and no guillotine breaks (complete separation) have occurred. This latter type of failure is unlikely to occur unless it can be shown that the duplex stainless steel castings in actual components have embrittled to such an extent that a brittle fracture could occur at or near the operating temperature.

10.6 In-service Inspection Methods (ISI)

The currently required in-service inspection (ISIs) have not always been able to detect cracks that have developed during service. Detection of IGSCC using the currently available ISI techniques is particularly difficult. Examination of cast stainless steel by ultrasonic testing is considered unreliable. The USNRC is currently sponsoring research into improved methods for continuously monitoring pressure boundary components.[10] On-line acoustic emission is being considered, but a question exists as to its ability to judge the severity of a particular cracking event. Proposals have also been made to install moisture-sensitive tape on the outside surface of critical piping to detect leaks.[11]

10.7 Summary, Conclusions, and Recommendations

The degradation sites and mechanisms, stressors, potential failure modes, and ISI methods for the BWR recirculation piping are summarized by Table 10.2. IGSCC has been a major problem and a variety of fixes and controls (discussed above) are in place. Other degradation mechanisms are possible and should be evaluated, but to date they have not been associated with actual outages or failures in BWR recirculation piping systems.

As might be expected, the nuclear industry is extremely optimistic about the possibility of extending the life of nuclear power plants, at least from the technical point of view.[12,13] The predominant view is that no insurmountable technical obstacles currently exist that would preclude the extended use of existing BWR plants, provided the groundwork is laid now to support continued use. The careful monitoring and recording of component performance from all operating plants is a key factor in this effort. The actual degree of thermal embrittlement in the duplex stainless steel castings should be monitored. The power industry, through industry groups like the Electric Power Research Institute, and the nuclear steam supply system manufacturers through programs like General Electric's COMPASS computerized data base and Westinghouse's Reliability Data Base, are attempting to develop useful data bases that will pinpoint weak links and identify trends useful in the life-extension effort. These data bases will be essential in the effort to ensure and demonstrate the safety and reliability of each plant being considered for life extension.

Information on failure mechanisms and sites should continue to be gathered from individual utilities, nuclear steam supply system manufacturers (such as General Electric), and reviews of USNRC reportable incident records to update the data base of known weak links and material failures. Additional work needs to be performed to analyze the existing data bases for trends in component failure mechanisms and locations. For instance, failure locations may be one of the keys in verifying the existence of so-called fatigue hot spots in BWR recirculation systems. The existence of such hot spots and their effect on overall system life becomes of greater concern as the number and severity of startup/shutdown cycles increase.

Additional work also should be done on duplex stainless steel castings to determine whether (a) a BWR environment will produce significant thermal aging and (b) typical castings contain sufficient delta ferrite to exhibit a significant embrittling effect.

The possible detrimental effects of hydrogen additions to the BWR recirculation piping loop should be studied to determine what reactions occur between hydrogen and other system components such as feedwater piping.

Table 10.2. Summary of degradation processes for BWR recirculation piping

Rank	Degradation Site	Stressors	Degradation Mechanisms	Failure Modes	ISI Method
1	Weld heat-affected zone furnace sensitized safe ends	Tensile stress, oxygen environment, sensitized heat-affected zone	IGSCC	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape
2	High thermal stress regions predicted by stress rule index analysis	Cyclic tensile stress, corrosive environment	Thermal fatigue, corrosion fatigue	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape
3	Austenitic-ferritic stainless steel castings with high delta ferrite levels	High temperature, tensile stress, shock loads	Thermal embrittlement	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape, impact test specimens
4	Shaft sleeves, other crevices	Corrosive environment, oxygen-starved areas	Corrosion	Material wastage, leaks	Moisture-sensitive tape

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11. NONDESTRUCTIVE EXAMINATION METHODS

L. J. House

Safety and reliable performance typically are achieved during the design life of nuclear reactors by using an engineering approach based on overdesign and incorporation of generous safety factors. Therefore, the demands on methods for detecting and measuring flaws using nondestructive examination (NDE) are modest. Historically, many of the American Society of Mechanical Engineers (ASME) code requirements and standard NDE methods were developed for detection and qualitative assessment of fabrication related defects generally associated with the quality of workmanship and then adapted to in-service inspection requirements within the normal design life of the components. On the other hand, reactor plant-life extension beyond the normal design life involves engineering calculations for cases where safety factors may be reduced because of in-service degradation of critical reactor components. Thus, life-extension inspection generally requires greater detection reliability and a more quantitative assessment of defects and accumulated damage than traditional fabrication and in-service inspection. Successful application of residual-life prediction programs, therefore, depends on the availability of adequate methodologies to detect and measure the effect of accumulated damage on performance-related properties of the key components.

The objective of this review is to address the efficacy of standard NDE methods for detecting and evaluating damage produced by degradation mechanisms occurring in reactor environments. It is well known that the reliability and capability of NDE methodologies are directly related to operator performance factors[1] and specific characteristics of the instrumentation, data acquisition, and signal processing employed.[2-4] However, this review is intended to be of a general nature covering the standard methodologies rather than attempting to assess the many specific embodiments of the techniques derived from those methods. Also, the adequacy of NDE for any given application depends on many other factors besides the performance of the technique. These factors include cost-benefit analysis (e.g., does inspection cost more than periodic component replacements?), availability of trained manpower, accessibility, and inspection rates. Only the technical-performance-related factors (i.e., can the methodology reliably provide the necessary measurement?) are addressed in this review.

The methodologies generally practiced by the nuclear and other industries and included in this review are visual examination, penetrant and magnetic particle testing, x-ray radiography, eddy-current testing, ultrasonic testing, and acoustic emission monitoring. These standard NDE methods are discussed with respect to their adequacy for providing the information about accumulated damage in reactor components necessary for residual-life estimation. The format for this review consists of a background discussion of what residual-life estimation requires of NDE, the type of information standard NDE methods can provide, and a summary of the general deficiencies in the standard methods for use in residual-life estimation.

11.1 Background

In the transition from the task of providing a simple quality control function to supplying quantitative assessment of accumulated damage for fracture mechanics analysis and residual-life estimates, the reliability and adequacy of standard NDE methodologies have rapidly diminished. Many of the standard NDE methods were intended to provide only an indication of the presence of a flaw or developed specifically to detect a particular type of flaw in a given component during fabrication and assembly. For residual-life estimates, those same methods must be employed not only to detect flaws but to determine the location, size, shape, orientation, and type of flaw or damage in components with various service histories, complex geometries, and limited access. The inspection is often performed in harsh environments with very severe constraints on the time available to do the inspection. Consequently, a number of inadequacies in standard practices of NDE have, and are continuing, to appear.

To estimate residual life using either a deterministic or probabilistic approach, information about the effect of various degradation mechanisms on performance-related properties is needed. As illustrated in Figure 11.1, the approach for each critical component has been to identify the principal degradation sites, establish the associated damage mechanism, determine performance-related physical and material properties affected by the damage mechanism, select the appropriate measurement technique to detect the relevant property changes, and finally to assess the extent and

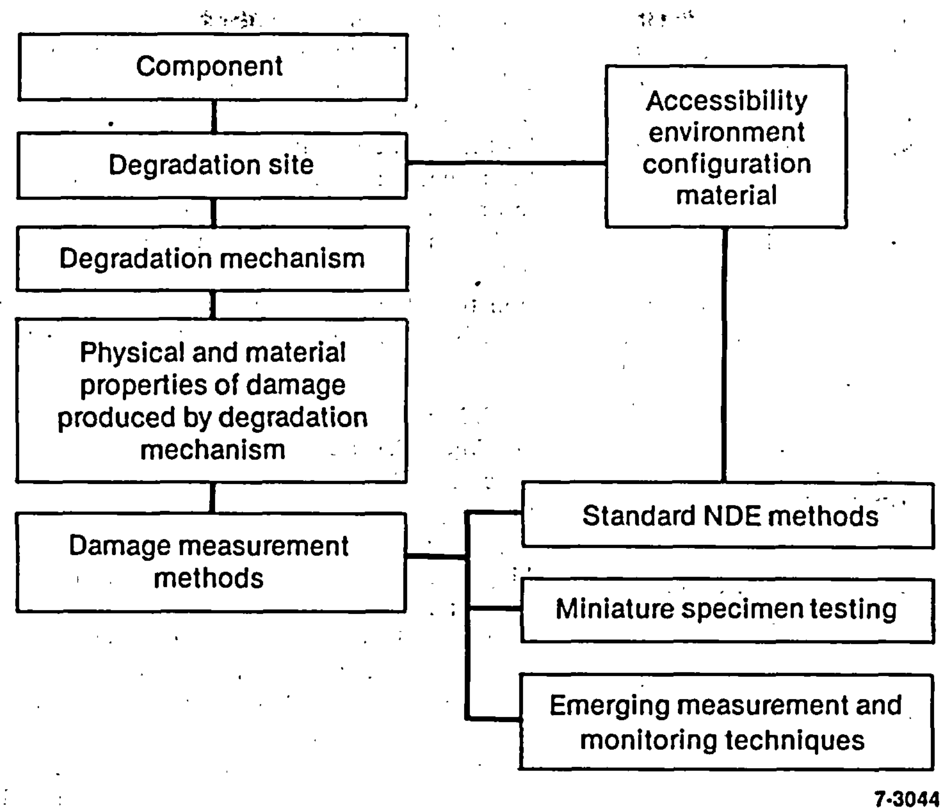

Figure 11.1. Typical approach to selecting a method for detection and measurement of accumulated damage in critical reactor components.

severity of the damage.[5-12] The selection of the measurement methodology may include standard NDE, miniature specimen testing, or one of the many emerging measurement and monitoring technologies. The selection of the appropriate method depends on accessibility, environment, component configuration, and the material at the potential degradation site.

The properties affected by degradation, particularly those detected and measured by standard NDE methods, are generally only indirectly related to the actual damage. For example, in the early stages of plastic deformation, there is a marked increase in dislocation density and associated stacking faults. This causes a decrease in the electrical conductivity of the material. Eddy-current methods can measure changes in conductivity and, thus, in principle can be used to measure deformation-related properties, such as yield strength. However, other factors, including variations in chemical composition, temperature, surface condition, and unrelated degradation mechanisms, such as thermal aging, also can affect conductivity. Also, changes in the concentration of interstitial solid solution hydrogen may distort the crystal lattice of the metal alloy, causing a change in its electrical conductivity. Therefore, the other sources of conductivity changes preclude this approach for most practical applications. Thus, if the NDE method can not discriminate between the types of damage or if a knowledge of the degradation mechanism causing the damage is unavailable, then the nondestructive measurement—based on empirical correlation of a physical property like electrical conductivity with a particular type of material degradation such as hydrogen damage or plastic deformation—can give unreliable results. This issue becomes very complicated because of the large number of potential damage mechanisms that exist in the reactor service environment.

The principal degradation mechanisms in key reactor components include: embrittlement (neutron, hydrogen, and thermal), fatigue (corrosion, thermal, low cycle, high cycle), corrosion (erosion, wastage, pitting, denting, and stress corrosion cracking), wear and fretting, and aging (thermal and environmental). The issue for residual-life assessment is whether standard NDE methods can detect the accumulated damage produced by these various mechanisms and provide a measure of the severity and extent of the damage in a sufficiently quantitative manner. For some of the degradation mechanisms, like neutron embrittlement, it is unclear what to measure using NDE methods in order to detect and characterize the resulting damage. Ideally, however, NDE should be able to discriminate between or identify the type of damage, measure the flaw size or extent of damage, measure the flaw shape and orientation, and determine the location of the flaw or damage. Although standard NDE has many deficiencies in being able to provide all of these necessary measurements for residual-life estimates, the information NDE can provide, despite the attendant uncertainties and reliability issues, is nevertheless essential.

11.2 Standard NDE Methods

Summarized in this section is the type of information provided by each of the standard NDE methods. More specifically, the flaws and damage that can be detected, the properties that can be measured, and the deficiencies of each method with respect to residual-life assessment are discussed.

11.2.1 Visual Examination.[13] Inspection by visual examination is the most widely used nondestructive method. Visual examination is important for reasons other than its simplicity and low cost. First, visual examination provides a qualitative indication of surface conditions. In welds, visual examination can reveal the presence of cracks, surface porosity, and other potential sources of mechanical weakness, such as the extent of penetration, undercutting, sharp notches, or misalignment. Furthermore, visual examination may assist in the determination of other nondestructive tests that should be applied to a potential site of degradation in a reactor component and how they should be applied. As with all nondestructive methods, the usefulness of visual examination methods depend on proper application and correct interpretation. In visual examinations, more directly so than in the other nondestructive methods, the results are subjective and dependent on the operator. The results of visual examination are generally qualitative and limited to assessing damage that manifests itself at an accessible surface of a component. Visual aids such as optical microscopy to examine the microstructure of a prepared surface and a borescope to examine the interior walls of tubing increase the utility of this method. In tube bundles, borescopic inspection provides information about the restriction of flow because of scale and deposit build-up and wall thinning caused by wastage, pitting, or uniform corrosion. The development of computer-enhanced vision systems to automate visual examination and to permit remote visual examinations in hostile or otherwise inaccessible environments have been successful for viewing structures, but the information provided is often qualitative in nature and of limited value to residual-life estimation. Visual examination in most cases provides an indication that damage may have occurred but cannot directly quantify the amount of material damage.

11.2.2 Penetrant and Magnetic Particle Testing. Penetrant and magnetic particle testing are essentially an extension of visual examination in that they are used as an enhancement technique to improve the visibility of surface-connected flaws. Magnetic particle testing also can indicate the presence of near-surface defects, such as cracks and inclusions. Its principal utility as a NDE method is to increase the inspection rate and probability of detection of surface flaws, compared with unaided visual examination. Liquid penetrants delineate surface discontinuities making the inspection less subject to the visual acuity of the operator. They provide no direct information about the depth of the crack although some very experienced inspectors have been able, in some cases, to make fairly reliable estimates of the depth and size of defects. The accuracy of these estimates depend on the magnitude of the defect opening at the surface being related to the depth of the defect. Penetrant methods can be used to locate grinding cracks, welding cracks, casting cracks, fatigue cracks, shrinkage, blowholes, seams, laps, pores, cold shuts, porosity, lack of bonding, pinholes in welds, through cracks, forging laps, bursts, gouges, tool marks, and die marks, provided the defects open to the surface. In some work with radioactive penetrants,[14] residual activity of the penetrant has been successfully correlated with crack size but complete capillary diffusion of the penetrant into the flaw is essential. In general practice, the correlation is not a reliable indication of crack size. The deficiencies of penetrant and magnetic particle testing are the same as with visual examination, except that the detection probability for surface-connected flaws is improved over unaided visual examination.

11.2.3 Radiographic Examination. The standard practice in radiographic examination is generally x-ray film radiography. The method essentially provides a recording of density variations that may occur because of a variety of causes: cracks, inclusions, porosity, voids, lack of bonding, and dimensional changes. The radiograph can provide quantitative information about flaws that affect the mass density or x-ray absorption coefficients of the component; for example, it can indicate the length of cracks. A film densitometer also can give an estimate of the actual extent of damage given suitable calibration and precise control of exposure time and source intensity. However, two-sided access, except in those few cases where autoradiography (a technique using the intrinsic radioactivity of the component) is possible, the slow rate of examination, background radiation, and the time between the examination and when the results are available after film development and evaluation severely restrict the application of this method. This is particularly true for periodic in-service inspections of reactor components where, to minimize costly down-time, the NDE activity must not delay decisions to replace or return a component to service.

11.2.4 Eddy-Current Testing. The use of single- and multifrequency eddy-current testing is generally limited to inspection of simple geometries for near-surface defects. The only extensive application of this method in the nuclear industry has been to inspect pressurized water reactor steam generator tubing. Although the eddy-current signal is affected by the metallurgical condition of the material being tested and can provide an indication of degradation or damage, the effect of changes in phase and chemical composition, as well as other metallurgical properties, on the eddy-current signal is neither quantitative nor predictable. A correlation between the measured impedance change of the eddy-current coil and the desired structural or serviceable characteristics must be established for all of the various degradation mechanisms. Normal metallurgical or chemical variations can mask the effects produced by service-related damage in the component structure and any condition affecting the permeability or conductivity of the material being tested affects the measured eddy-current signal. Discriminating between changes in the measurement caused by normal material variations or actual damage is often not possible. Single-frequency eddy-current examination of steam generator tubing in accordance with the ASME code procedures has not always been reliable because, in addition to the reasons cited above, other inspection variables, such as tube supports, tube sheets, denting, and eddy-current probe wobble, affect the results. Multifrequency methods remove the influence of many of these variables, but improvements are still needed in the sensitivity and reliability of flaw detection and characterization. One particular deficiency is that standard eddy-current methods and probes for tubes and methods are inadequate for detecting and characterizing circumferential type flaws, like intergranular cracks and general intergranular attack, in tubes in the neighborhood of and within the tube-sheet crevice. The design of pancake coils for tube inspections enables the detection of circumferential flaws, but the use of this type of probe is not presently a standard practice for tube inspections.

11.2.5 Ultrasonic Testing. After visual examination, the next most widely used volumetric nondestructive method for in-service inspection of components is ultrasonic testing. The ASME code requires ultrasonic inspection of welds and adjacent base material. Although presently accepted as the most useful volumetric examination method, the reliability of standard ultrasonic procedures and methods of flaw detection has been increasingly questioned during the last several years.[3,15-17] This is because the ultrasonic examination methods, particularly procedures and practices specified by the various ASME code requirements, were developed primarily to assess overall quality by detecting fabrication-related defects that correlated with workmanship.[16] However, the information required for residual-life assessments and fracture-mechanics-based integrity assessments requires detecting, locating, and sizing of more subtle flaws. Sizing of flaws has always been a source of error because ASME code sizing techniques are typically based on empirical correlations of ultrasonic-echo-amplitude measurements with flaw size and not on basic physical principles.

The standard ultrasonic examination methods generally use recording of pulse-echo amplitude and transducer position as the basis for flaw detection and sizing. The functional aspects of ultrasonic testing involve two operations. The first operation is generally a rapid initial search mode to detect the presence of a flaw in a component and a second operation in which a more detailed inspection is performed to determine whether the flaw is benign or represents a performance-limiting condition. Standard procedures, although satisfactory for their original purpose, are not adequate and are often unreliable for providing the detection and characterization of flaws needed to assess the residual life of reactor components. Some issues contributing to the inadequacy of standard ultrasonic testing practices include theoretical limits of detectability and resolution; external factors such as operator training, equipment, and calibration deficiencies; acoustic coupling variabilities; and internal factors such as material variabilities, part geometry, and flaw characteristics. An example of internal factors that affect flaw-detection reliability is the variability of ultrasonic echo strength with mechanical load. A mechanical load of sufficient magnitude can compress the two faces of a crack tightly enough so that the ultrasonic reflection is reduced or, in the extreme case, the crack becomes transparent. Rough-clad weldments and components fabricated from cast stainless steels are virtually uninspectable by standard ultrasonic methods. Other deficiencies include unreliability of detection of cracks under cladding, detection and sizing of intergranular stress corrosion cracking (IGSCC), and detection of thermal fatigue cracks and IGSCC in wrought stainless steel welds.

It is clear that the limitations of standard ultrasonic test methods are in part caused by deficiencies and physical limitations of the available technology. On the other hand, a previous evaluation of the reliability of Air Force field inspections indicated that inspection results were poorer than could be explained on the basis of the inherent physical limitations of the technique.[15,18] Poor performance was attributed to human factors such as operator boredom and the awkward positions in which the operator was required to perform the inspection, because of limited access.

A good summary of the capabilities of standard ultrasonic NDE methods is given in Figure 11.2,[3] in which diagrams of the probability of defect detection (DD), correct rejection (CR), and correct acceptance (CA) as a function of defect categories are shown. The defect categories included—small acceptable defects (a), small rejectable defects (b), large continuous defects (c), and composite (multiple-type) defects (d). The ultrasonic test procedures fell into three categories. Category A complied with the requirements of the Program for the Inspection of Steel Components (PISC)—formerly Plate Inspection Steering Committee—which closely follow the requirements of ASME XI. Category B involved minor modifications to the PISC procedure, and Category C was markedly different ultrasonic techniques providing truly alternative procedures. Only the large continuous defects are properly detected and rejected by the Category B and C ultrasonic test procedures. The Category C procedures do result in better, but not perfect, small-defect detection.

11.2.6 Acoustic Emission Monitoring. Acoustic emission measurement techniques have been used extensively as an alternative to ultrasonic examination for detecting flaws in pressure vessels in the nuclear and petrochemical industry. Acoustic emission is a passive technique that detects flaws only during growth. One very useful application of acoustic emission monitoring is leak detection in pressure vessels. A major deficiency in acoustic emission testing is the possibility of missing a large flaw if it happens to not be growing during monitoring. Advocates of acoustic emission testing counter with the argument that only flaws that are growing can induce failure and continuous acoustic emission should preclude missing the large flaws during growth. A number of major evaluation programs have been performed on pressure vessels with mixed results. Difficulties are associated with discriminating between damage- or flaw-related acoustic emission events and ambient background noises, low-level emissions produced by flaw growth in certain alloys, and accurate location of the acoustic emission event in complex structures. An intrinsic deficiency of standard acoustic emission monitoring is the absence of a reliable correlation between measurement of acoustic emission events and the severity of the flaw. A present advantage of most acoustic emission monitoring is its use as a precursor to impending failure rather than a method for quantitative measurements of component damage for use in fracture-mechanics estimates of flaw severity or residual-life assessment . Another advantage of acoustic emission monitoring, unlike the previous methods discussed, is that a more global examination or large-volume interrogation is provided. A major disadvantage of acoustic emission testing for residual-life estimation at present is the absence of a correlation between acoustic emission properties and fracture-mechanics-related parameters. For example, acoustic emission parameters can be used for crack-growth detection or location, but the measured parameters have not been reliably correlated with crack size.

11.3 Summary of General Deficiencies of Standard NDE Methods

NDE methods, as embodied in standard practice, offer capabilities that are primarily limited to flaw detection. Standard NDE methods are presently

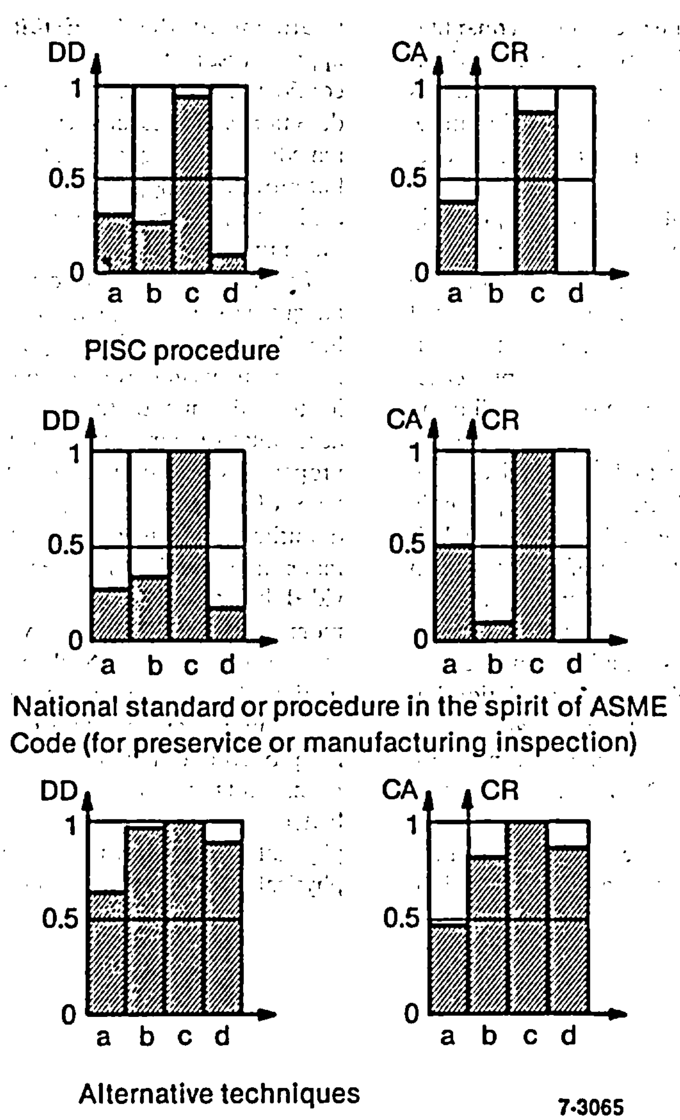

Figure 11.2. Average cumulative diagrams of defect detection (DD) and correct rejection or acceptance [CR(CA)] as a function of the defect categories a, b, c, and d (from Reference 3).

measurements of physical properties that, based on empirical correlation, are related to defects and component damage. In other words, most NDE measurements are only indirectly related to the flaw size or damage state. These measurements are often affected by geometry and material properties unrelated to damage and flaws. Changes in test conditions may produce entirely false indications and some comparisons of damage state using different inspection methods or simply different inspection times may be invalid. Having to rely on flaw calibration standards and empirical correlations also may make it prohibitively expensive to develop NDE methods for all types of damage, materials and heat treatments, component geometries, and the various service histories of each component.

The high cost of down time and slow inspection rates (area or volume of material inspected per unit time) associated with the high-resolution techniques necessary for detecting small critical flaws and providing flaw characterization represents another deficiency of standard NDE methods. Global inspection methods, even with low resolution, are needed to initially determine whether a more detailed inspection is warranted and to minimize NDE bottlenecks in returning a system online. Acoustic emission is one of the few NDE methods offering considerable potential for providing a global inspection approach.

Residual-life estimation and fracture-mechanics analysis require more than flaw detection. Ensuring structural reliability when extending components beyond their original design life necessitates a quantitative determination of the effect of accumulated damage on the performance related properties of various components. When a NDE operator is asked to make these types of measurements, an entirely different set of problems arise than those encountered in simple flaw detection. Standard NDE methods are presently incapable of handling this new set of problems.

Eddy-current and ultrasonic methods are among the most promising for making the type of damage-related measurements needed for residual-life estimation, but models for quantifying the effect of material properties and associated degradation mechanisms are still being developed. Presently, there is poor agreement between models and experiments in actual structural components because models have been well developed only for idealized defects in infinite media. There are deficiencies in the present understanding of the interaction mechanisms between the various manifestations of performance-related structural damage and nondestructive measurements. This requires considerable empirical correlation to establish a reliable relationship between the presence or extent of damage and a nondestructive measurement. The constraints of in-service inspection limit most standard NDE to providing only qualitative assessments.

One approach for handling the uncertainties in the reliability of flaw detection and sizing is a probabilistic treatment of the NDE data. The probabilistic approach estimates residual life with a specified level of confidence given the uncertainty associated with the detection and measurement of flaws. This does not remedy the inadequacies of standard NDE methods but provides an approach for assessing the risk associated with relying on the information that NDE can presently provide.

The ultimate goal is to develop an NDE capability adequate for a more quantitative evaluation of residual life. Realization of this goal requires significant improvements in flaw detection reliability and the ability to make quantitative measurements of the effect accumulative damage has on performance related properties of a component. Discrimination between types of flaws, accurate quantitative flaw sizing, and quantitative measurement of performance-related material properties is presently not adequately provided by standard NDE methods. Furthermore, a more fundamental deficiency affecting the development of the necessary NDE methods for reliable estimation of residual life of components is related to the limited understanding of the relation between the physical properties of the material, the degradation mechanisms, and their effect on component performance. A better understanding of the effect of damage on these physical properties is needed in order to identify the physical measurements that need to be made.

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12. ADEQUACY OF ASME CODE IN-SERVICE INSPECTION METHODOLOGY

J. F. Cook

Federal regulations[1] require that the requirements of Section XI of the American Society of Mechanical Engineers (ASME) Code[a] for in-service inspection (ISI) of nuclear power plant Class 1, 2, and 3 components and systems be used. The edition and addenda of Section XI that must be followed also are specified in the regulations. The code edition/addenda to be used for the ISI is determined by the date of the start of plant operation and is normally updated whenever a new addenda is issued. However, the updating of ISI plans to later editions/addenda can be delayed by extended shutdowns. Therefore, the code edition/addenda followed varies among the operating plants. This is further discussed in Reference 2. Because there is no one set of current ISI requirements, this chapter discusses the general philosophy of the ISI specified in Section XI, and the status of improvements in the ASME code requirements relative to selected components important to plant safety and plant life extension. Advanced NDE methodologies are also discussed.

12.1 ASME Section XI Code Rules

The acceptance criteria in ASME Section XI for flaws found by nondestructive examination (NDE) are based on fracture-mechanics analysis. This is in contrast with the fabrication code (Section III) NDE acceptance criteria that are based more on workmanship standards. However, the NDE requirements initially put in Section XI were based on the same requirements used in the past for the fabrication code NDE. This has resulted in an incompatibility between the Section XI acceptance standards and the NDE capabilities. In many cases, the NDE is not capable of detecting the Section XI acceptance limit indications. For example, incidents of through-wall intergranular stress corrosion cracking (IGSCC) have been found in piping systems by evidence of leakage that were not detected by standard ASME code ultrasonic inspection. Studies performed under the Program for the Inspection of Steel Components[3] (PISC) (formerly the Plate Inspection Steering Committee) on heavy-section specimens showed inadequacies in standards, ASME code minimum flaw sizes, and vessel ultrasonic examination procedures. Improvements have been made and work is in progress to correct some of these limitations. Those improvements and work in progress are discussed below as related to certain components identified as important to life extension.

The Section XI nondestructive examinations serve as a complement to pressure tests specified in Section XI that must be applied to all safety-class pressurized components. The pressure tests act as temporary stressors providing for: (a) some verification of continuing structural integrity, and (b) identification of flaws that cause leakage. Unlike NDE, which is generally applied only to welds and adjacent base metal, the pressure tests check the complete pressure boundary. Also, the NDE is done on a sampling plan while all components receive the pressure test. However, these pressure tests only evaluate the integrity of the pressure retaining boundaries; they do not check other components important to safety, such as supports, and they do not detect small defects that may later grow to a critical size.

The NDE specified in Section XI is done on a sample of the welds. For example, 25% of the reactor coolant piping system welds must be examined during each ISI interval. If flaws are found, then additional examinations are required. Also components containing flaw indications are required to be examined more frequently in the future (e.g., reexamined during the next three successive inspection periods). The more frequent examinations are designed to determine whether the flaws are growing. Examples of these requirements are contained in Sections IWB-2420 and IWB-2430 of Section XI.

A code requirement that needs to be resolved is the inspection schedule for times beyond the present 40-year plant life. Two different inspection programs are presently allowed by the code. One program (Inspection Program B) uses four 10-year intervals. Inspection Program A uses, from the start of operation, 3 years for the first interval, 7 years for the second interval,

a. When used in this Chapter of this report, unless otherwise stated, references to the Code or Section XI means *Rules for In-service Inspection of Nuclear Power Plant Components*, Section XI, Division 1 ("Rules for Inspection and Testing of Components of Light-Water Cooled Plants") of the ASME Boiler and Pressure Vessel Code.

13 years for the third interval, and 17 years for the fourth interval. Most plants now use Program B. This is fortunate because Program A reduces the ISI frequency as the plant operates longer and, thus, would give less ISI information for assessing plant condition relative to life extension.

The code has historically specified NDE requirements that included details on how to perform the examinations (a cookbook approach). The philosophy of Section XI of the code is now changing to specify NDE performance standards, leaving the procedures up to the performing organizations. The performance demonstration requirements are intended to verify that the NDE system (equipment, procedure, and personnel) is capable of detecting the minimum size flaws identified in the Section XI acceptance standards. Additional performance demonstrations are being required for flaw sizing. Examples of these requirements for piping welds are contained in a draft of Code Case N-409, Revision 1, now under consideration by the Section XI ASME Code Committee. Implementation of the performance demonstrations should result in more uniformity in the performance of NDE systems (procedures, personnel, and equipment) for flaw detection and flaw length and depth sizing.

12.2 NDE Improvements

Reference 4 contains extensive information on the degradation of bolting in nuclear plants. Partly as a result of the problems that are described in this reference, the first code document that requires NDE performance demonstrations (Section XI, Appendix VI, "Ultrasonic Examination of Bolts and Studs") was included in the Winter 1983 Addenda. The licensee is required to demonstrate that the procedures and personnel can detect a qualification notch representing the maximum acceptable size flaw. Data on field experience using these requirements should be available following USNRC reference of this code addenda in the regulations and subsequent update of plant ISI plans.

Piping ultrasonic examination practices have received great attention in the code because the problems with IGSCC in boiling water reactor plants have increased. As a result of the IGSCC of certain larger pipes, first discovered at the Nine Mile Point Plant, two USNRC Inspection and Enforcement (I&E) bulletins have been issued.[5,6] These bulletins require qualification of piping examination procedures and personnel on blind specimens containing IGSCC. These practices were subsequently covered by Code Case N-409.[7] Additional ASME code changes that are being developed may add performance demonstration requirements for all piping weld ultrasonic examinations. This activity will significantly improve the effectiveness of ultrasonic examination practices for finding flaws in piping welds. However, there are still problems in developing effective ultrasonic procedures that apply to all piping. The geometry of welds and adjacent fittings often prevent the access required for conducting a complete ultrasonic examination.[8] In many cases, even special ultrasonic techniques will not penetrate cast stainless steel piping and fittings.[9]

Changes were published in the Winter 1983 Addenda to Section XI that simplified selection of, and reduced the number of, Class 2 welds to be included in examination samples. This change also improved examination methodology in that all surface-only examinations were eliminated. Data collected from piping failures show that most problems (e.g., IGSCC and thermal fatigue cracks) originate from the inside surface. Similar changes should be made to Class 1 examination requirements. At present, Class 1 piping with less than a 4-in. nominal pipe size receives only a surface examination. Confirmatory data on the effectiveness of ultrasonic examinations on small piping would help in obtaining code committee approval of changed Class 1 requirements.

Improvements in vessel ultrasonic examination requirements are presently being acted on by the ASME Boiler and Pressure Vessel Committee. It is common practice now for most foreign and domestic plants to apply additional examinations beyond the code minimum. As a result of international cooperative studies on ultrasonic vessel examination practices known collectively as PISC and recommendations from the Pressure Vessel Research Committee, requirements for an additional examination (70-degree examination angle) for near-surface flaws (e.g. underclad cracks) are being added to the code. In this same action, caution is being added to the code language regarding using amplitude/search unit movement techniques for flaw sizing. The Section XI code action in progress is scheduled to be published as Appendix I to Section XI. Another significant improvement in this code action is a requirement for more sensitive recording that will reduce the chance of missing flaw indications.

Containment structure leakage rate testing is controlled by Appendix J of the USNRC regulations. As well as the leak testing, a general visual examination of accessible internal and external surfaces is required. However, minimum accessibility requirements are not given. Thus, the extent of the examinations probably varies among the plants. Subsection IWE of Section XI of the code contains rules for the examination of metal containment structures. USNRC action to require use of these rules is in progress. When implemented, these subsection IWE requirements, which are concentrated on welds and other important parts of the containment structure (e.g., bolting, seals, and gaskets), should result in more consistent metal containment examination practices. Most of the examinations are visual and conducted on painted surfaces. Reference 10 indicates that visual examinations can appropriately detect degradation in painted metal structures.

Rules for concrete containment examinations are now approved for Code publication as Subsection IWL of Section XI in the Winter 1986 Addenda. Additional tests that could enhance the new subsection IWL rules relative to life assessment for concrete containments are given in Reference 10. These tests, controlled by British standards, include checking carbonation depth, ultrasonic pulse velocity, covermeter, and rebound hardness measurements. These tests will be reviewed and compared with the American Society for Testing and Materials practices. Following this review, recommended changes will be presented to the ASME for improving concrete containment examination practices relative to plant-life extension. Previous work at the Idaho National Engineering Laboratory involved correlating ultrasonic pulse velocity measurements with changes in concrete properties following exposure to radiation.[11] Reference 10 emphasizes the importance of destructive tests for determining life forecasts of concrete structures.

The Section XI steam generator tubing eddy-current examination requirements are based on the single-frequency methodology published in Regulatory Guide 1.183.[12] However, most examinations are now performed using multiple-frequency methods, which can enhance detection of flaws in areas such as those adjacent to tube-support structures. The code has not yet required that the eddy-current procedures be qualified for detection of cracks in tubing. Calibration using flat bottom holes (simulating wall thinning) is specified. The eddy-current response to a flat-bottom hole is usually significantly different than the eddy-current response to a crack. Other steam generator examination practices that are now being used by industry such as ultrasonic testing of tubing, profilometry measurements and radiography are scheduled to be covered by the ASME; work has been started at the task group level of the ASME code committee. Further discussion of steam generator examination practices is included in Chapter 6 of this report.

The Section XI NDE requirements are focused on finding discrete flaws in components; detection of general degradation (e.g. radiation damage, thermal aging) of material properties is not covered per se. However, control specimens for destructive testing are specified for inferring reactor vessel material properties after periods of operation. As part of checking materials for aging relative to life extension, it may be desirable to have material property data on other components such as piping sections, supports, pumps, etc. Samples of these components are not now available for destructive tests. However, miniature specimen techniques (see Chapter 14 of this report) may prove useful in this regard. Also, some of the required properties such as yield strength, tensile strength, and ductile-to-brittle transition temperature may be inferred from NDE data using advanced techniques, as reported in References 13 through 17. These references indicate that measurements of ultrasonic attenuation and velocity can be used to infer certain mechanical properties because the mechanical and ultrasonic properties are all dependent, to a degree, on material microstructure. Measurement of changes in the mechanical properties because of environmental factors, (such as thermal aging and radiation damage) should be possible for a specific material once calibration data for the material have been obtained. The results to date have been obtained on laboratory test specimens; application of the techniques to actual components in the field must await perfection of ultrasonic instrumentation and data processing methods now being developed that will allow measurements where only one material surface is accessible and will minimize surface and ultrasonic coupling variations. Further confirmatory tests and field demonstrations will be required before any of these techniques can be included in the ASME code. Addition of material property NDE requirements to Section XI would significantly improve the knowledge of the actual in-service properties, and also might reduce the need for testing surveillance specimens.

12.3 ASME Code ISI and Life Extension

The current Section XI ISI requirements can provide only part of the information needed for making decisions regarding life extension. Several weaknesses in NDE practices need to be resolved. For example, in most cases, the code does not require any backup examinations to confirm results from the primary examination. However, selected utilities are applying backup examinations. If an ultrasonic examination finds indications of cracks in a pipe, it may be desirable to use an inspection port to go inside the pipe with a remote visual or surface examination to backup the ultrasonic examination. Geometric conditions in components sometimes produce indications that are confused with indications from flaws. It is common practice to always specify a backup examination for the United States Air Force nondestructive inspection programs. Backup examinations will increase the confidence in NDE and provide better information on the condition of components relative to plant-life extension.

Improvements are being made in the NDE methodology to enhance the reliability of flaw detection. After a flaw is found in a component, the flaw must be characterized (sized) to determine the effect of that flaw on the life of that component, or else the component must be repaired or replaced. Experience with the IGSCC problems in piping has shown that qualifying equipment and personnel for sizing is more difficult than qualifying for detection. Applying standard ASME code rules for sizing of flaws in vessels can in reality end up giving the size of the ultrasonic soundbeam. Corrective action is underway as shown in Supplement 12 of Appendix I of the code. A significant problem in developing flaw sizing requirements will be defining the qualification specimens. The economics of fabricating an adequate number of vessel specimens will slow the code action on this problem. However, life-extension considerations may increase interest in solving this problem.

12.4 Summary, Conclusions, and Recommendations

ISIs of nuclear power plants are controlled by the USNRC regulations and Section XI of the ASME code. Improvements in the ASME code NDE methodology relative to flaw detection are evolving and are summarized in Table 12-1. These improved code rules should be applied as part of evaluating plant conditions relative to license renewal. Current code methodology is not wholly adequate to assess the residual life of the major light water reactor components. More efforts are needed to develop field-usable nondestructive examination techniques and equipment that can provide valid information on the sizing of defects and the material properties of the major components.

Table 12.1. Summary of code NDE improvements

Component	NDE Method[a]	NDE Improvements	Status
Bolting	UT	Demonstration of procedure and personnel qualification	Published, W-83 Addenda
Piping	UT	Procedure and personnel qualification using statistically based specimen sets, containing flaws	In committee, may be in W-87 Addenda
Vessels	UT	Near surface examination requirement, correction of flaw sizing techniques	In committee, may be in W-86 Addenda
Containments	VT, LT	Addition of Code examination rules to supplement the Appendix J leak testing	NRC review in progress
Steam generators	ET, RT, UT Profilometry	Code rules to cover all types of NDE used by the industry	In committee

a. UT, VT, LT, ET, and RT refer respectively to ultrasonic, visual, leak testing, eddy-current, and radiograph nondestructive examination methods.

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13. CURRENT LIFE ASSESSMENT TECHNIQUES

C. E. Jaske

Both experimental and analytical techniques are used to assess the remaining life of components that are in-service. These include testing of surveillance specimens, monitoring of operational parameters, evaluation of samples removed from service-exposed components, and prediction of damage accumulation processes. In actual practice, one or some combination of these techniques typically are used in making residual-life assessments.

Surveillance programs are used to assess the influence of neutron irradiation on the mechanical properties of materials used in the beltline region of reactor pressure vessels (RPVs). Owners of light water reactors (LWRs) are required by federal regulations to maintain such programs, and American Society for Testing and Materials (ASTM) Designation: E 185 defines the "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which provides guidelines for a minimum surveillance program. Samples of base metal, weld metal, and heat-affected-zone metal are enclosed in surveillance capsules. The capsules are placed within the RPV at locations such that the amount of irradiation leads is greater than that at the inside surface of the beltline region by a factor of 1 to 3. A schedule for capsule removal and sample testing is established so that the long-term effects of irradiation can be assessed. Both tension and Charpy V-notch tests are required. Other tests, such as hardness and fracture toughness, are optional.

The status of surveillance programs in the United States has been described in detail elsewhere.[1] The integrated surveillance program used for Babcock & Wilcox plants has been described by Lowe.[2] Data from the surveillance programs provide indications of the effects of radiation on tensile strength and Charpy-impact properties. Figure 13.1 shows the correlation of brittle-to-ductile transition temperature shifts at 30 ft-lb (41 J) with tensile yield strength. A similar correlation of fractional decrease in upper-shelf energy is shown in Figure 13.2. These two figures are from the work of Odette,[3] and they show that consistent trends exist. The major shortcoming of such data is that their relationship to more relevant properties for fracture-mechanics analyses, such as fracture toughness and crack-arrest resistance, is not well established.

Operational parameters, such as temperatures, pressures, and amount of cycling, can be monitored to provide an assessment of remaining life. These parameters are not direct indicators of the material's condition or state, so some type of damage accumulation algorithm is needed to estimate their impact on material degradation. For example, if thermal fluctuations were monitored at a critical nozzle region, one would have to compute the corresponding thermal strain history in that region and then use an appropriate fatigue-damage accumulation model to calculate estimates of expended life and remaining life. A practical example of such on-line monitoring of the condition of an oil-fired 1000-MW boiler is presented by Davidson.[4] As recently pointed out by Toman,[5] equipment monitoring techniques should be (a) nondisruptive, (b) noninvasive, (c) repeatable, (d) accurate, and (e) cost-effective, in addition to yielding data that are correlatable with the deterioration process. As an example, he discusses the monitoring of the condition of an emergency diesel-engine generator system for a nuclear power plant.

Another approach to life assessment is to directly measure the properties of samples that have been removed from components. Typically, a boat, core, or ring sample is removed from the component, and the component is then repaired using procedures that satisfy the applicable regulations and safety codes. As will be discussed later in Chapter 14 of this report, there has been some recent interest in removing miniature samples to either eliminate or greatly reduce the amount of required repairs. The main advantage of this approach is that it provides a direct measure of the mechanical properties of the material after service exposure. However, archival data for that same material are not usually available for comparison and assessment of the degree of degradation. Using typical properties for that comparison and assessment adds to the uncertainty of the analysis. Another problem with this approach is that the tests must be accelerated to obtain information in a reasonable time period during an outage. If a time-dependent degradation mechanism, such as corrosion or corrosion-fatigue behavior, is being evaluated, the results of the accelerated testing must be extrapolated to make remaining service-life predictions, and the extrapolation introduces additional uncertainty.

As an example, the direct properties measurement approach was used by Battelle to provide data for assessing the structural integrity of Type 316

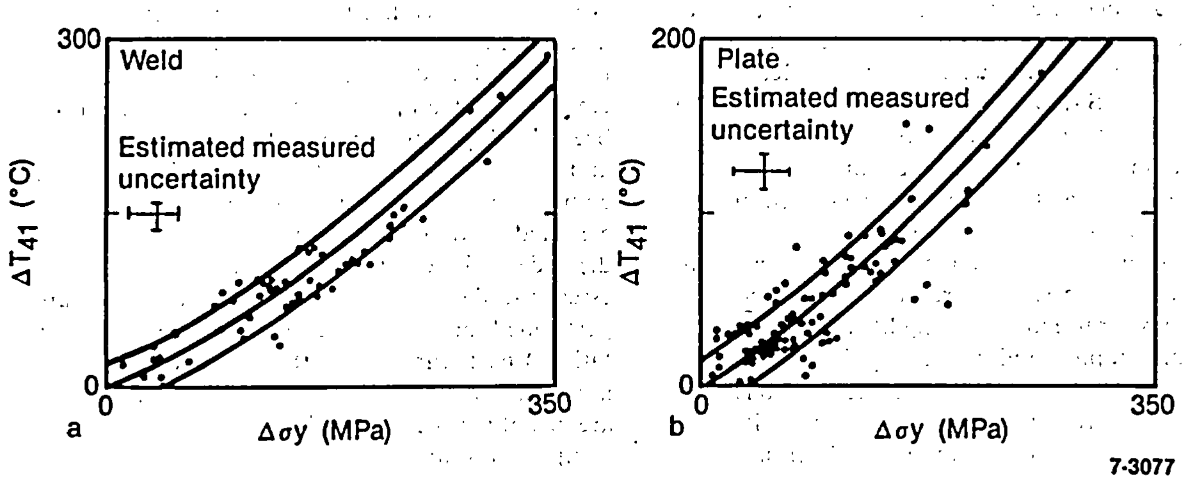

Figure 13.1. Plots of transition temperature shifts indexed at 30 ft-lb (41 J) versus static yield stress changes for (a) weld and (b) plate and forgings.

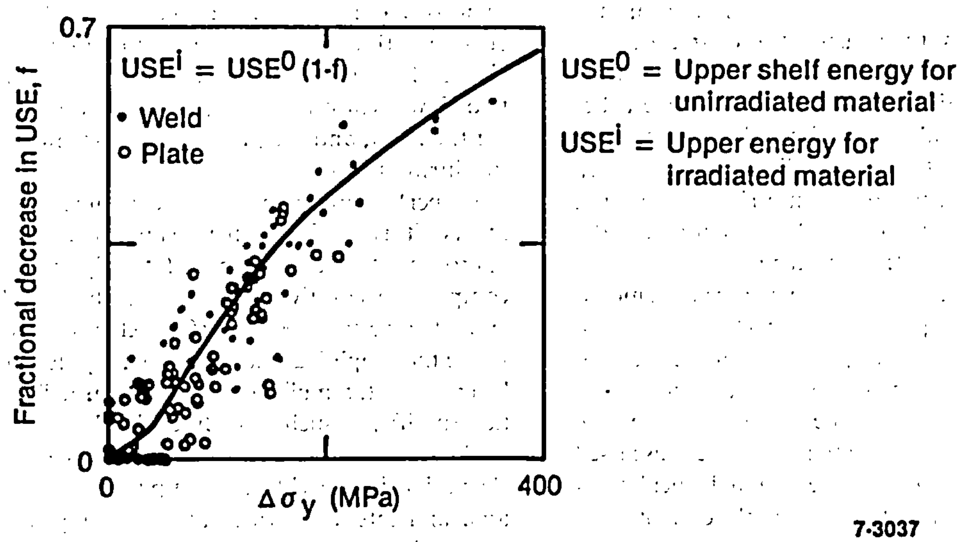

Figure 13.2. Fractional decreases in C_v upper-shelf energy versus yield stress changes.

stainless steel weld metal from the Maine Yankee reactor coolant system.[6] Fatigue test bars were removed from boat samples of weld metal that contained microfissures. Low-cycle fatigue specimens were machined from those bars and tested using standard procedures comparable to the test procedures that are used to develop fatigue data for the establishment of American Society of Mechanical Engineers (ASME) code design fatigue curves. The fatigue data then were used to evaluate structural integrity in terms of the expected fatigue usage factors and the safety factors incorporated in the ASME code. As another example, tensile, fatigue, fracture toughness, and Charpy specimens have been taken from steam-turbine shafts to measure properties for use in remaining life analysis.[7]

Metallographic and fractographic examination of samples from in-service components also is used to help assess remaining life. For example, if volumetric nondestructive evaluation (NDE) has indicated the presence and growth of a defect or both, a sample of material containing the indicated defect may be removed and sent to a laboratory for detailed examination using conventional metallurgical techniques to ascertain the nature and extent of the defect. Suspected regions of intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) piping systems have been examined in this manner. In fossil plants, ring samples of superheater and reheater tubes are often examined metallographically to measure the degree of fireside and steam-side corrosion and the amount of creep damage.[8] Similar samples of furnace tubes from petrochemical plants have been examined metallographically to measure the amount of creep voids and fissures, where an empirical correlation has been established between those factors and the service-failure rate of the furnace tubes.[8]

The analytical prediction of damage accumulation is an important technique used in remaining life assessment. In fact, some type of prediction of the accumulation of damage during anticipated future operations is required to make any estimate of remaining life. The models of material degradation processes discussed previously are used to make the required calculations of damage propagation starting from the current damage state of the material.

If no measure of the current damage state is available, one can estimate it by calculating the damage accumulation during the past operating history. However, knowledge of the past operating history and the condition of the material that was originally put into service is often incomplete, adding a great deal of uncertainty to this calculation-of-current-condition approach. Thus, some destructive or nondestructive measurement of the current condition of the material is preferred for remaining life analysis.

Analytical techniques for the prediction of remaining service life have been pioneered by the aircraft industry, where NDE and a fracture-mechanics-based, damage-tolerant approach is employed.[9] Complex computer programs have been developed to compute fatigue-crack growth and assess the potential for fracture in structural components throughout the course of various expected loading spectra. Because of the history dependence inherent in material degradation processes, the damage accumulation calculations are performed on event-by-event (cycle-by-cycle for fatigue) basis throughout the course of the simulated loading. For this reason, the computations are usually time-consuming and demanding of computer resources.

The type of analytical procedure described by Jaske[8] is often used to assess the remaining life of steam-turbine rotors. A flow diagram showing the technologies used in performing the life-assessment analyses is presented in Figure 13.3. Examples of the results are presented in Figure 13.4. Figure 13.4 (a) shows the ultrasonic indications that are obtained from boresonic inspection of a rotor. Cluster analysis is used to group the indications in regions that are considered to be single flaws in the flaw-growth analysis. The remaining parts of that figure show the calculated stress distributions at two flaw locations, the expected loading spectrum for the rotor, and the predicted flaw-growth behavior during future service.

In performing plant-wide remaining-life assessments, it is useful to have simplified, rule-based approaches that can be applied rapidly and easily to compute conservative life estimates. Then the detailed and extensive analyses are performed only for the cases where the simplified rules show that a potential problem may exist. Simplified methods are usually application specific, so such procedures should be developed for key reactor components, such as RPV welds, nozzle welds, piping welds, coolant pumps, diesel generators, and internals, where the possibility of fatigue damage is of concern.

Engle,[10] for example, has developed a simplified approach for estimating fatigue-crack growth in typical aircraft structural components under spectrum loading. Figure 13.5 shows a comparison of predicted versus experimentally measured fatigue-crack growth for several design limit stress levels that were made using his simplified approach.

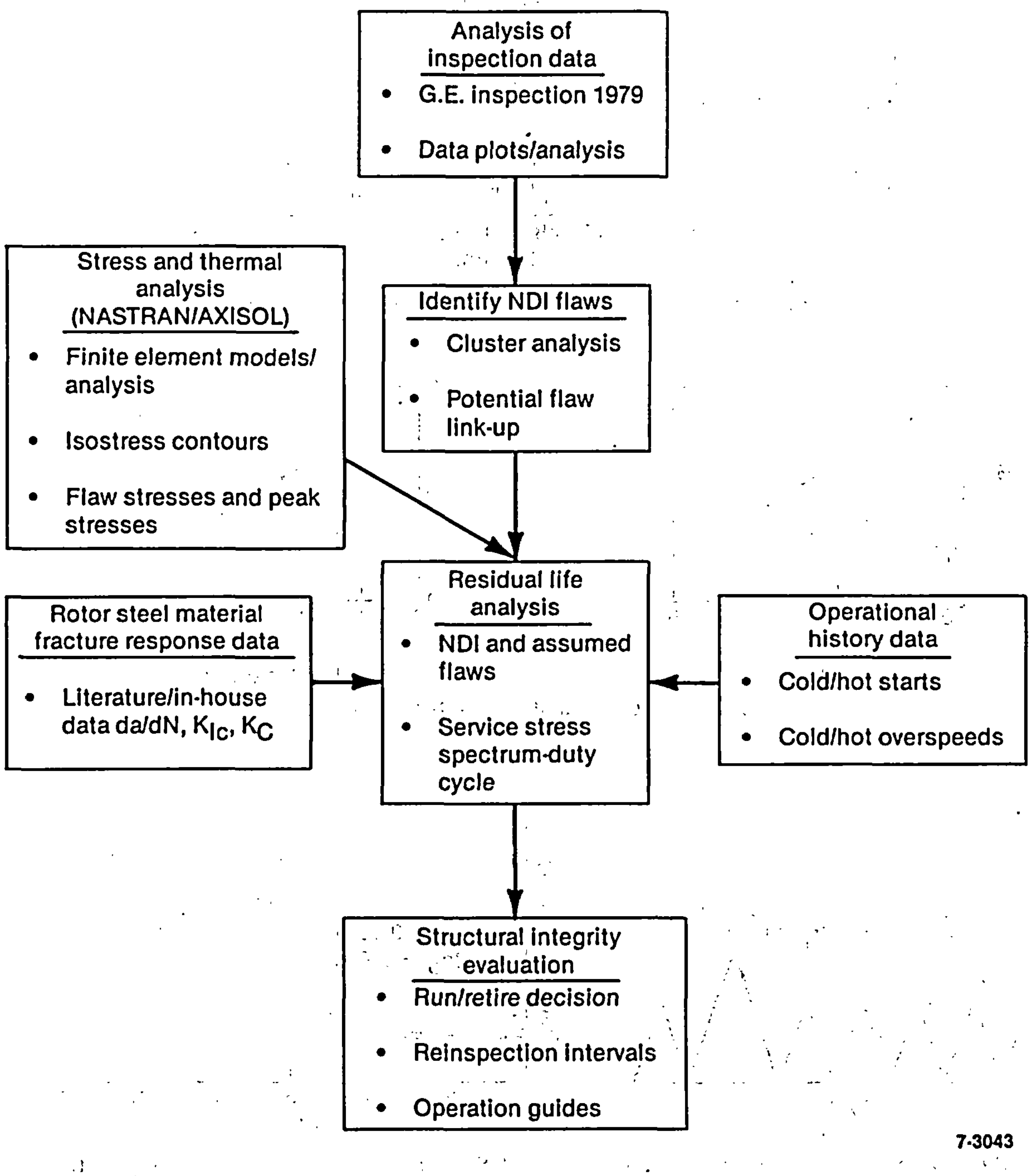

Figure 13.3. Life assessment analysis approach for steam-turbine components.

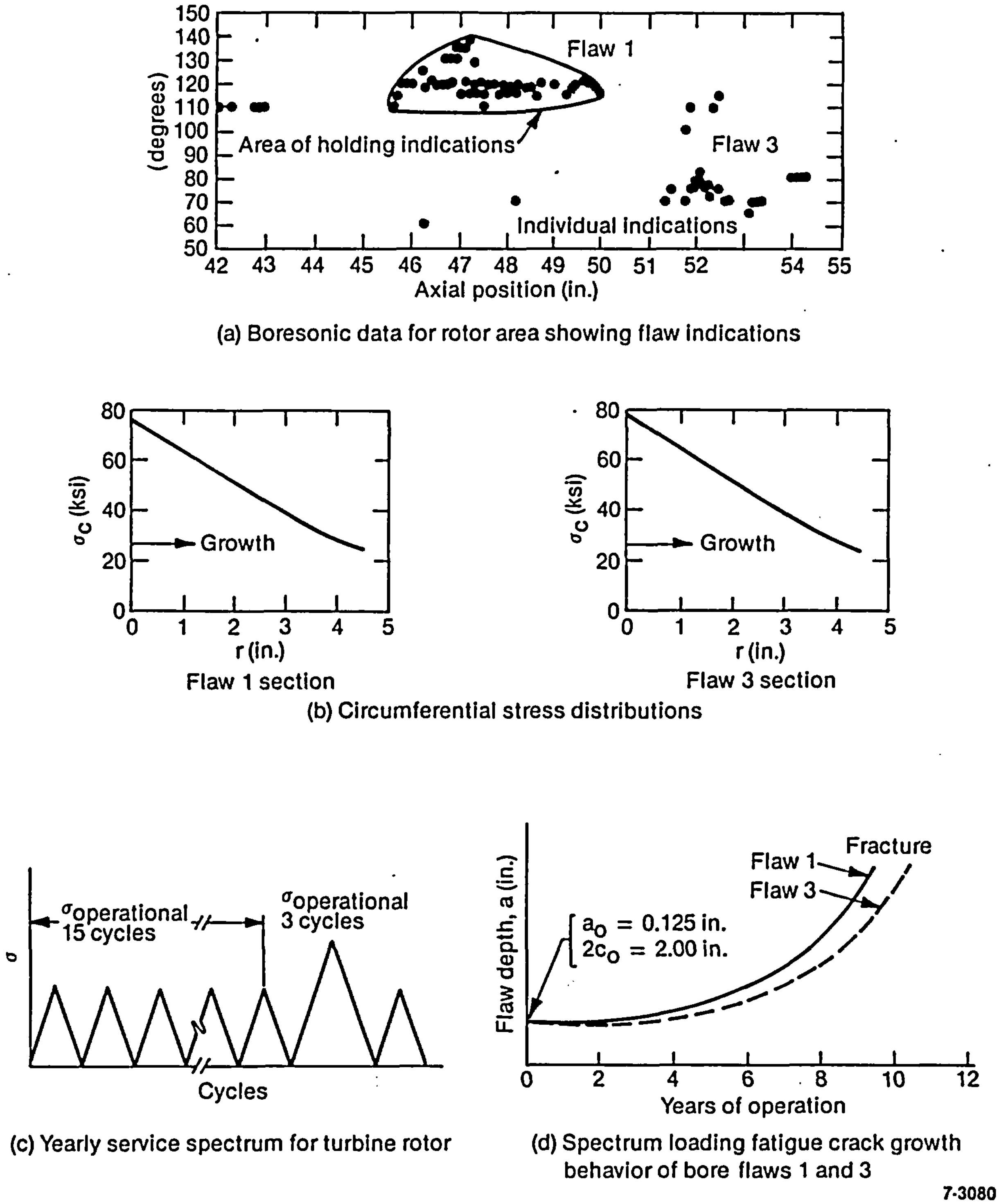

Figure 13.4. Example of results from a life assessment of a steam-turbine rotor.

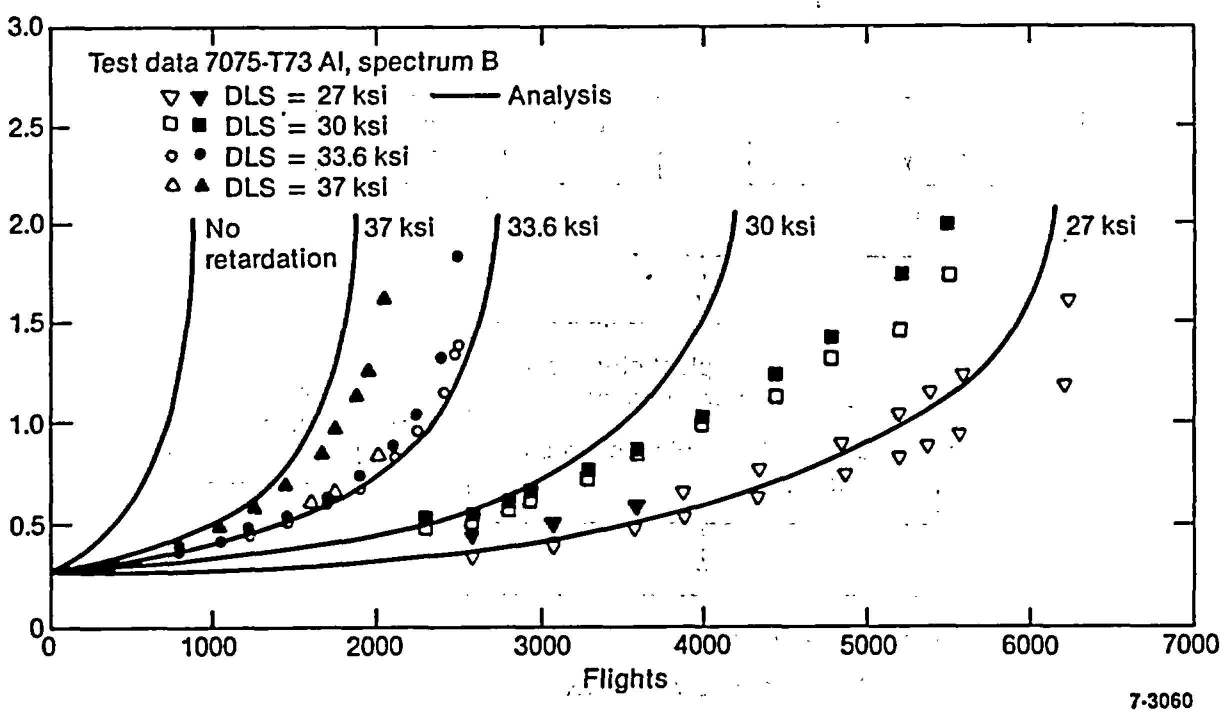

Figure 13.5. Correlation of predicted with measured fatigue-crack growth rates under spectrum loading.

Saxena et al.[11] applied fracture-mechanics technology to predict the residual life of a ships service turbine generator (SSTG) casing using the general approach illustrated in Figure 13.6. The region of concern had a complex configuration as is shown in Figure 13.7 and was subjected to cyclic thermal stresses. They concluded that casings with cracks <6.3 mm deep could be operated safely for another 800 cycles, and they recommended intervals for future periodic inspections. This same type of residual-life analysis could be applied to nuclear reactor components with appropriate modifications to the materials properties, configurational models, and failure criteria.

Broek[9] has summarized the six basic types of information that are needed to apply a fracture-mechanics approach:

- The minimum detectable flaw size and type
- A prediction of the residual strength of the structure with a flaw present and of the critical flaw size associated with the fail-safe load
- A description of the expected future loading history
- A determination of the crack growth response from the minimum detectable size to the critical size associated with failure
- A knowledge of the critical locations in the structure where cracks are likely to develop, such as geometric discontinuities associated with fatigue-loaded welds
- A reliable inspection of the various important regions of the structure or component, including consideration of the accessibility to those regions.

With current NDE methods, defects can be detected with relatively high confidence. However, as discussed in chapters 11 and 12, the sizing and classifying of those defects has much lower confidence and is an area where major improvements are needed and where much NDE research is concentrating.

Predictions of residual strength and critical flaw size can be made with reasonable confidence for brittle fracture where linear elastic fracture mechanics can be applied. For nuclear pressure boundary components, however, the possibility of ductile fracture is often of concern. Inelastic fracture mechanics methods for dealing with ductile fracture problems in piping and pressure vessels are still evolving. Those methods need to model leak-before-break and crack-arrest, so that fail-safe conditions occur at the critical crack size. For applications to actual components, these methods need to be developed and verified for dynamic as well as static loading conditions.

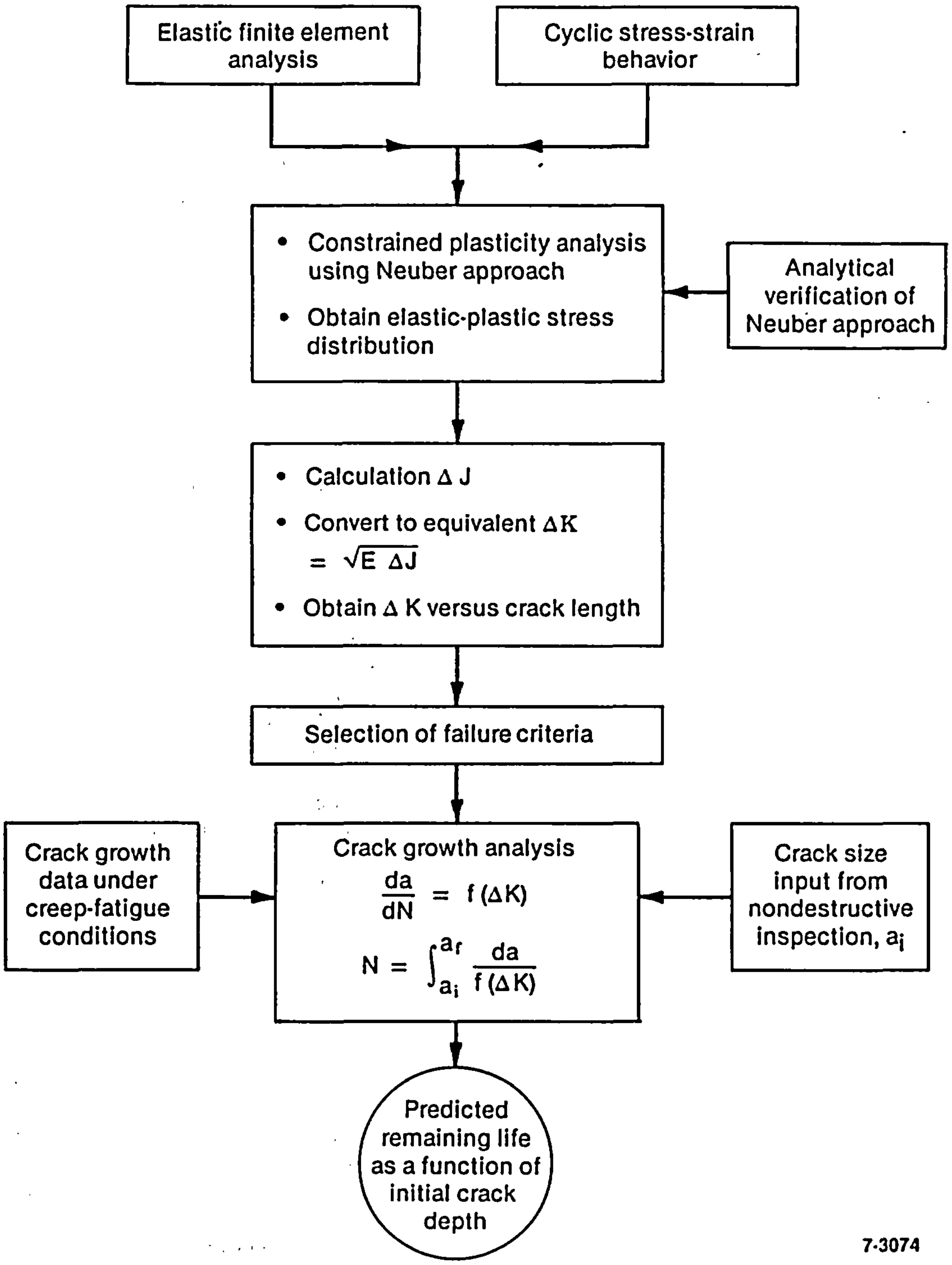

Figure 13.6. Methodology for remaining life prediction of SSTG casings.

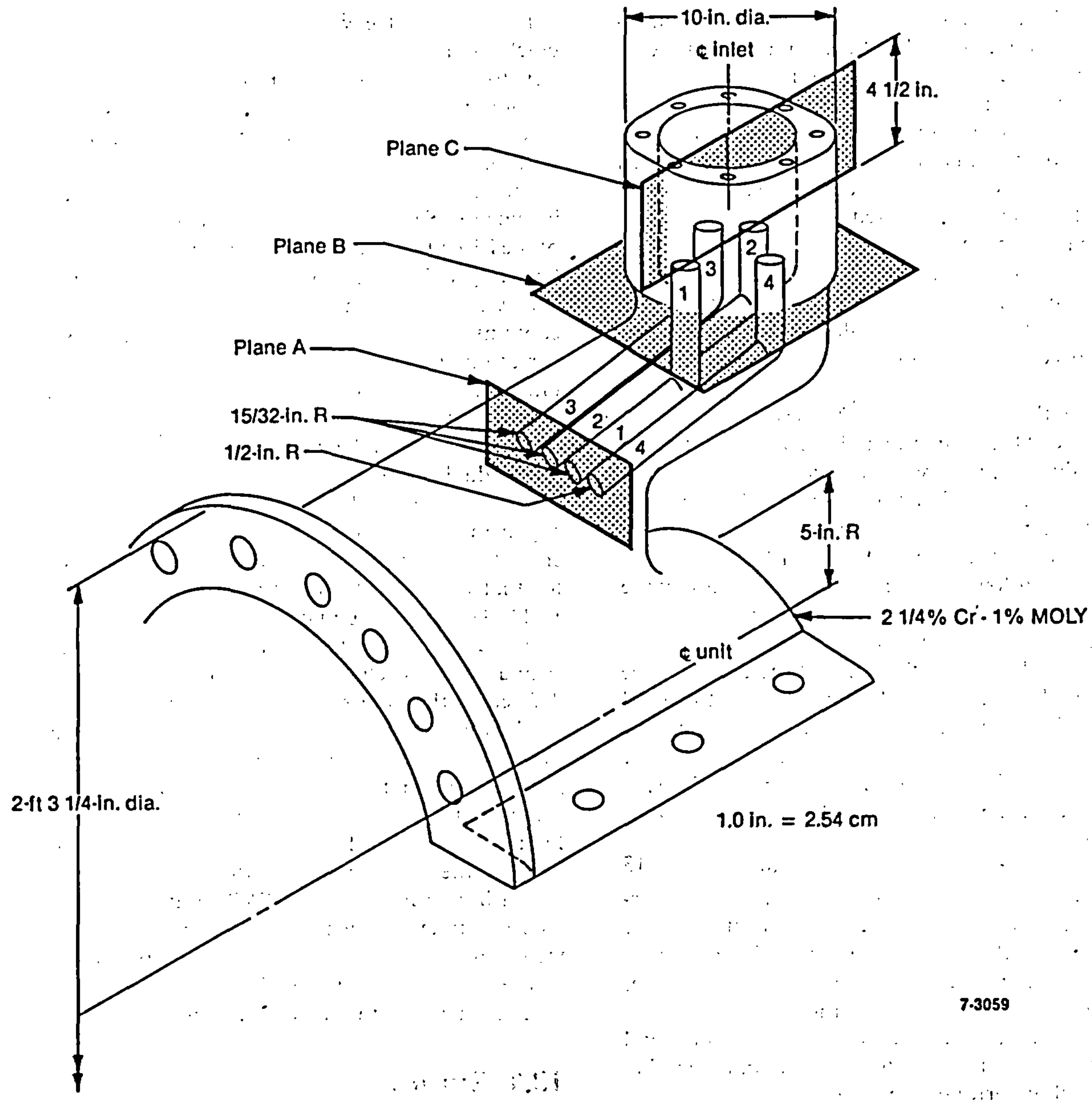

Figure 13.7. Illustration of configuration of region where remaining life of SSTG casing was assessed.

Descriptions of expected loading histories are obtained from past experience and from postulated worst-case scenarios that may be encountered. Because damage accumulation is strongly history dependent, it is important that such loading histories be well defined. Unfortunately, as pointed out by Yahr et al.,[12] this is not the case in the nuclear industry, so they had to develop a unit histogram for the Zion-1 plant for use in their fatigue damage analyses. This type of work needs to be pursued so that sets of reference histograms can be developed for routine use in remaining life calculations.

The cyclic crack growth behavior of pressure vessel steels in LWR environments has been well characterized and that work is being extended to piping materials. One area of concern is at very low cyclic growth rates where there is some debate over the existence of a threshold for cyclic crack growth. Additional work is needed to resolve this question.

As discussed in the earlier chapters of this report, the areas where cracks are likely to develop in nuclear-plant components have been identified from operating experience and original design calculations. These include transition regions in the

RPV, closure studs, safe end welds, nozzle connections, elbows, control rod drive mechanism penetrations, and other welded regions in the pressure vessel and piping system.

Inspections of important regions of the nuclear-plant components are performed on a scheduled basis. Improved quantitative descriptions of detected flaws are required for use of a fracture-mechanics approach.

All six of the items needed to apply a fracture-mechanics approach are actually probabilistic in nature. Ideally, they should be quantified and employed in a probabilistic remaining life assessment methodology.

Recently, Sundararajan[13] has critically reviewed the state of the art in probabilistic assessment of the reliability of pressure vessels and piping. Most of the applications of that methodology have been in the nuclear power industry. Figure 13.8 shows the predicted failure rate of PWR RPVs for various reactor shutdown pressure-temperature paths at the end-of-life (EOL) fluence. For life extension, either the core loading patterns would have to be modified to prolong the attainment of the EOL fluence or the allowable pressure-temperature path would have to be modified to provide an acceptable risk of failure. In residual-life assessment, this type of analysis should be made on a plant-specific basis to allow for different materials properties and NDE results.

As is illustrated in Figure 13.9, Sundararajan[13] also has predicted the probability of protective barrier penetration from random fragment strike. If the effective thickness or the penetration resistance of the barrier is reduced during service exposure, then the probability of penetration is increased. For life extension, the barrier must be maintained and refurbished as required to maintain an acceptably low level of perforation probability.

Predictive methods for environmentally assisted degradation and cracking are not as well established as those for fatigue. A recent ASME symposium[14] was devoted to documenting the current state of the art in this technical field. The typical approach to solving corrosion and stress-corrosion cracking problems has been to adopt fixes that will eliminate or minimize the occurrence of those problems in the future.[15,16] That approach has been largely successful, so the predictive methods may not be required. However, in order to extend plant life, the components and regions of components that are potentially susceptible to environmentally assisted degradation must be identified by inspection so that the appropriate fixes can be implemented.

In the area of environmentally assisted fatigue cracking, a great deal of work has been devoted to developing predictive models[14,17,18] and Section XI of the ASME Code requires assessment of environmental effects on fatigue-crack growth. Bamford[19] has reviewed the basis for reference fatigue-crack-growth rate curves that are presently included in Section XI. Based on detailed reviews of the EPRI Database on Environmentally Assisted Cracking (EDEAC), revisions of those reference curves are being considered.[20] The EDEAC also contains data from slow-strain-rate and stress-corrosion-cracking tests, as well as the data from fatigue-crack-growth tests. Thus, it provides an important resource of very relevant materials properties data that can be used in developing methods for residual-life assessment.

Gilman[21,22] used EDEAC to develop a useful engineering model for predicting corrosion-fatigue crack growth rates in LWR environments. The basis for that model is illustrated in Figures 13.10 and 13.11. The lower bound of the cyclic crack growth rate in BWR water, which coincides with the upper bound of the cyclic crack growth rate in air, is defined as the base rate. The actual time rate of crack growth in LWR water then is correlated with the base rate. Figure 13.10 shows that there is a good correlation between the growth rates in BWR water and the base rate times frequency, and Figure 13.11 shows that there is a good correlation between the upper bound of the growth rates in pressurized water reactor (PWR) water and the base rate times frequency. These correlations provide a useful tool to help estimate the remaining service life in areas of the RPV where corrosion-fatigue crack growth is a potential problem.

13.1 Summary

The major life-assessment techniques available for application to LWR components are summarized in Table 13.1. For each technique, its characteristic features and its applications and/or potential applications are briefly highlighted.

Testing of specimens from surveillance capsules is used to assess the amount of radiation embrittlement that occurs in RPV steels during service and is an important technique for ensuring the structural integrity of the RPV at its EOL fluence. Monitoring of operating parameters to establish aging trends is not widely practiced, but with the development of appropriate baseline data it can provide a viable, economic tool to assist in

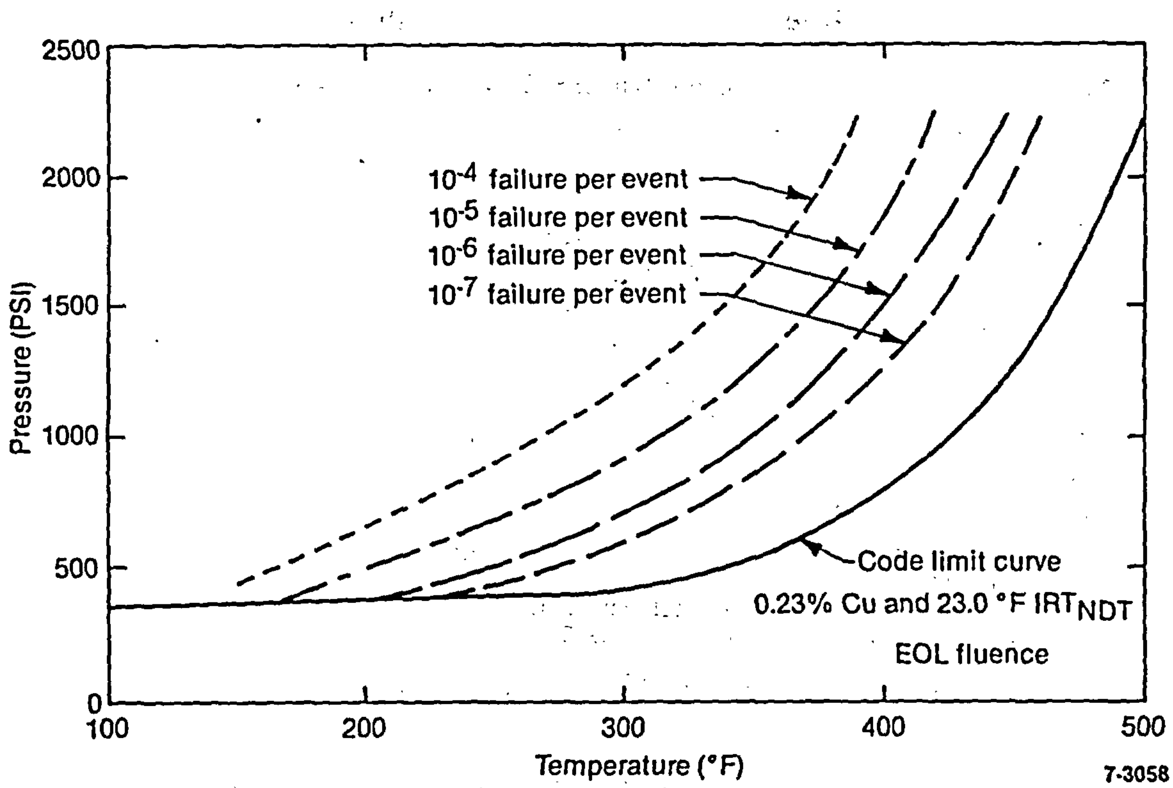

Figure 13.8. Constant failure rate and code-allowable pressure temperature paths for reactor shutdown at EOL fluence.

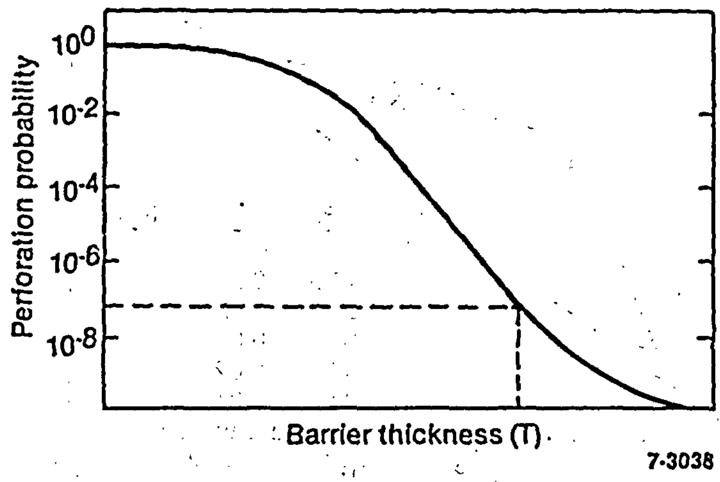

Figure 13.9. Probability of perforation versus barrier thickness.

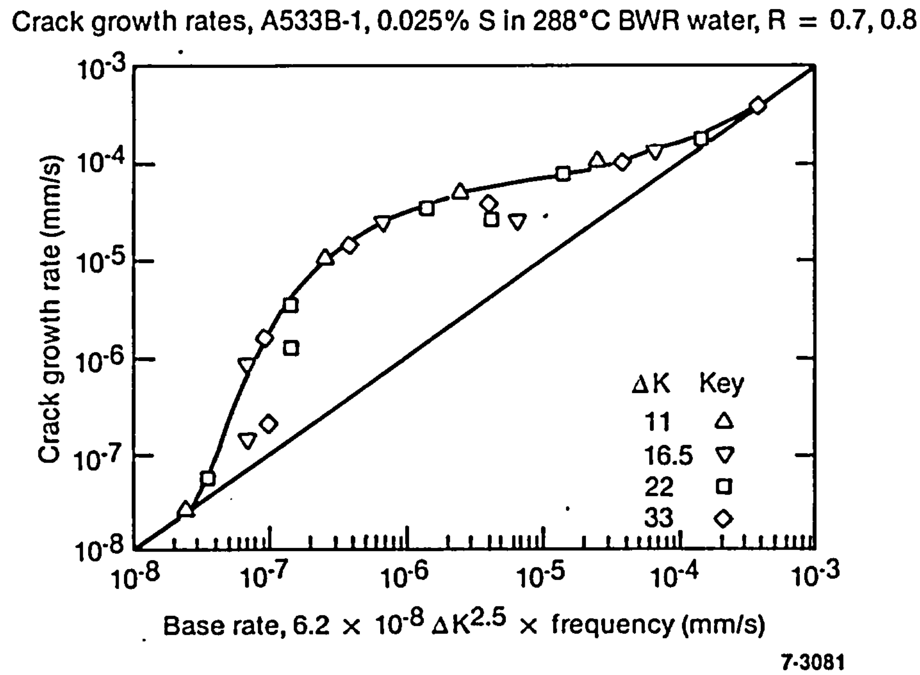

Figure 13.10. Time rate of fatigue-crack growth in BWR water as a function of base rate.

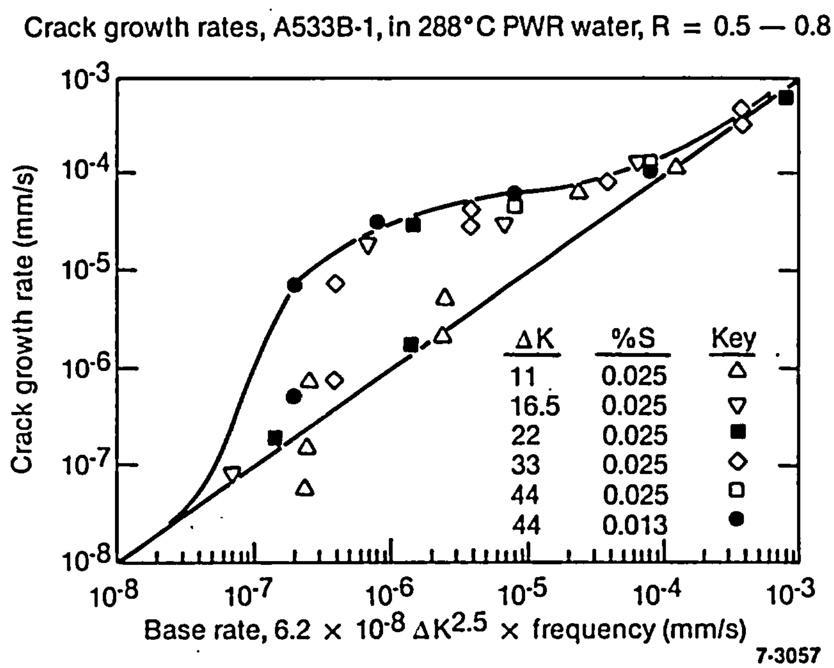

Figure 13.11. Time rate of fatigue-crack growth in PWR water as a function of base rate.

Table 13.1. Summary of current life assessment techniques

Technique	Characteristics	Applications
Surveillance programs	Uses specimens exposed to neutron irradiation in capsules within the RPV to measure tensile and impact properties.	Measuring the radiation embrittlement of RPV steels. Limited amount of test material is usually available. The relation of tensile and impact properties to fracture toughness is not well defined.
Monitoring of operations	Measure and record temperatures, pressures, and cycling for trending of performance.	Used to assess condition of mechanical equipment, such as pumps, bearings, engines, and generators. Need baseline data for normal operating conditions.
Testing of specimens removed from components	Measure mechanical properties of boat, core, or ring samples removed from the component.	Used on piping welds in regions where repair is possible. Not widely used because material removal is destructive. Need to develop methods for testing miniature specimens.
Metallographic and fractographic examination of samples removed from components	Measure metallurgical aging, void formation, and amount of microcracking. Identify mechanisms of cracking.	Used on piping welds in regions where repair is possible. Not favored because removal of material is destructive. Need to develop methods that use miniature samples or surface replicas.
Analytical prediction of damage accumulation	Compute the amount of damage accumulation expected during future operations.	Widely used for areas where high amounts of fatigue damage may occur. Need to develop realistic engineering approaches.

remaining life assessment. The destructive testing and examination of samples removed from components provides useful data, but is not useful for routine applications because repairs must be made.

Analytical predictions of damage accumulations are widely used in both design and remaining-life assessments. Current approaches typically use very simplified damage accumulation algorithms and large safety factors to ensure conservative predictions. More realistic approaches that allow the incorporation of history effects, fracture mechanics, and probabilistic factors need to be developed, especially to resolve difference between the ASME Code Section III and Section XI approaches to fatigue damage analysis.

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14. NEW OR EMERGING METHODS FOR INSPECTION AND LIFE ASSESSMENT

C. E. Jaske and L. J. House

As a direct result of the recent increased interest in extending the operating life of all types of major industrial plant equipment, including nuclear power-plant equipment, new methods for inspection and life assessment of that equipment are being developed. The inspection methods are required to evaluate the current damage state of the materials in such equipment without introducing a significant amount of additional damage from the measurement technique that is used. Current remaining life-assessment methods, such as those of Section XI of the American Society of Mechanical Engineers (ASME) Code, are based on inspection to find cracks or crack-like defects and using a fracture-mechanics approach.

14.1 Nondestructive Examination Techniques

The principal thrust of current research and development in nondestructive examination (NDE) is to extend the state of the art from qualitative flaw detection and estimation of the size of large continuous flaws to quantitative measurements of performance-related material properties and more accurate quantitative flaw characterization. Emerging NDE technologies consist essentially of variations and improvements in existing measurement or NDE methods rather than new sensor technologies.

The fundamental basis of all NDE is predominantly the measurement and analysis of either electromagnetic or mechanical radiation. Of the emerging technologies, ultrasonics continues to be the dominant source of new inspection techniques and advances in quantitative inspection technology.

The development of automated and expert systems for ultrasonic, eddy-current, and radiographic inspection also represents a major contribution to advanced NDE. Improvements in digital data acquisition of eddy-current and ultrasonic inspection data are now enabling real-time analysis and archiving of inspection results and permitting the application of adequately sophisticated models to provide more meaningful and quantitative interpretation of the inspection data. Advances in imaging systems also offer promise for greater realization of the potential of radiographic examination. Improvements in digital radiographic image processing include qualitative image enhancement and quantitative image analysis. The development of video systems with improved resolution presents the possibility of practical real-time radiography offering the advantages of immediate presentation of a flaw image and potentially enhanced flaw detection probability because many angles can be examined. If combined with digital processing, real-time radiography can provide a powerful quantitative inspection tool.

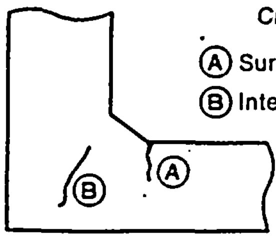
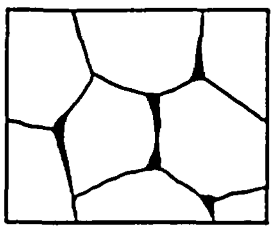
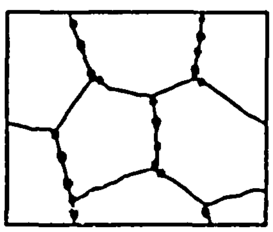
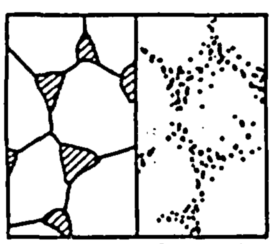
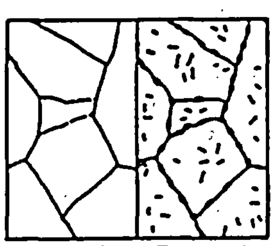
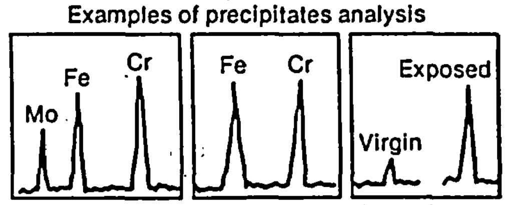
A shift from the development of periodic inspection methods to in situ monitoring techniques offers some advantages. Because monitoring techniques rely on detecting changes in material parameters rather than making absolute measurements, a number of calibration problems that limit the usefulness and reliability of periodic NDE techniques based on making absolute measurements are avoided. With in situ monitoring, the possibility arises for detecting microstructural or metallurgical changes in a material, measuring changes in component stresses, and measuring internal temperature. Of the techniques that provide large area monitoring capabilities, acoustic emission is one of the more promising emerging technologies. Advances in analytical models for more quantitative analysis of acoustic emission events, the development of small aperture arrays, and greater sophistication in the hardware for data acquisition offer considerable promise for acoustic emission to become a useful tool to monitor structural integrity.

Table 14.1 shows Broek's[1] summary of the seven basic nondestructive inspection techniques that were discussed in Chapter 11 of this report and are used in conjunction with the fracture-mechanics approach. The basic physical principles and the typical application are summarized for each technique. The major challenge for quantitative use of these techniques in a fracture-mechanics context is to accurately determine the size, shape, location, orientation, and type of crack or flaw. Furthermore, it would be useful to extend these techniques to other types of material aging or damage.

As is illustrated in Figure 14.1, Masuyama[2] summarizes methods that could be used for the NDE of materials that have been in service for some period of time, pointing out that both surface and internal cracks are of concern. Most of the traditional NDE methods can be used to detect surface cracks. However, only

Table 14.1. Inspection methods (from Broek)

Method	Principles	Application
Visual	Naked eye, assisted by magnifying glass, low power microscope lamps, mirrors.	Only at places easily accessible. Detection of small cracks requires much experience.
Penetrant	Colored liquid (penetrant) is brushed on material and allowed to penetrate into cracks. Penetrate is washed off and quickly drying suspension of chalk is applied (developer). Remnants of penetrant in crack are extracted by developer and give colored line.	Only at places accessible. Sensitivity of same order as of visual inspection.
Magnetic particles	Part to be inspected is covered with a layer of a fluorescent liquid containing iron powder. Part is placed in strong magnetic field and observed under ultraviolet light. At cracks, the magnetic field lines are disturbed.	Only applicable to magnetic materials. Parts have to be dismounted and inspected in special cabin. Also notches and other irregularities give indications. Sensitive method.
X-ray	X-rays emitted by (portable) x-ray tube pass through structure and are caught on film. Cracks, absorbing less x-rays than surrounding materials, are delineated by black line on film.	Method with great versatility and sensitivity. Interpretation problems if cracks occur in fillets or at the edge of reinforcements. Small surface flaws in thick plates difficult to detect.
Ultrasonic	Probe (piezo-electric crystal) transmits high-frequency wave into material. The wave is reflected at the ends and also at a crack. The input-pulse and the reflections are displayed on an oscilloscope. Distance between first pulse and reflection indicates position of crack. Interpretation: Reflections of cracks disappear on change of direction of wave.	Universal method because a variety of probes and input pulses can be selected. Information about the size and the nature of the defect (which need not be a crack) are difficult to obtain.
Eddy current	Coil induces eddy current in the metal. In turn, this induces a current in the coil. Under the presence of a crack the induction changes; the current in the coil is a measure for the surface condition.	Cheap method (no expensive equipment) and easy to apply. Coils can be made small enough to fit into holes. Sensitive method when applied by skilled personnel. Little or no information about nature and size of defect.
Acoustic emission	Measurement of the intensity of stress waves are emitted inside the material as a result of plastic deformation at crack tip, and as a result of crack growth.	Inspection while structure is under load. Continuous surveillance is possible. Expensive equipment required. Interpretation of signals is difficult.

Damage mode		Nondestructive detecting method
	Cracking (A) Surface cracks (B) Internal cracks	— For surface cracks • Penetrant test • Magnetic particle test • Ultrasonic test • Electric resistance method — For internal cracks • Radiographic test • Ultrasonic test
	Micro cracks	• Replication method • Physical test (not established) [Ultrasonic Electric Magnetic Radiographic]
	Micro voids	• Replication method • Physical test (not established) [Ultrasonic Electric Magnetic Radiographic]
 Virgin Exposed Ferritic steel	Structural degradation [Pearlite spheroidizing]	• Replication method • Extraction method • Physical test [Hardness X-ray diffraction] • Electro-chemical test
 Virgin Exposed Austentic steel	[σ phase precipitation]	Examples of precipitates analysis  Carbide σ phase Intensity depends on exposed time

7-3078

Figure 14.1. Methods for nondestructive examination of materials after service exposure.

volumetric methods, such as radiographic and ultrasonic testing, can be used to detect internal cracks. Metallographic replication can be used to detect microstructural damage, such as cracking along grain boundaries or voids, at accessible surfaces, but physically based NDE methods for the detection of such microstructural damage have not been developed for general use. Time-dependent changes in microstructural features, such as the ferrite phase in austenitic-ferritic stainless steels and precipitate phases in stainless steels, can be detected by replication, extraction, hardness testing, x-ray diffraction, and electrochemical testing at accessible locations.

Several general review articles on the current status and future prognosis of NDE technology have been published recently.[3-5] Posakony[3] pointed out that the future thrusts in NDE development include (a) defect characterization by means of advanced systems, (b) continuous monitoring of equipment by means of acoustic emission and vibration analyses, and (c) assessment of materials properties and monitoring of material condition by various NDE techniques for use in life extension efforts. McClung[4] indicated that improvements in NDE are coming from (a) the increased use of computers for acquisition, processing, and interpretation of large quantities of data in the field, while the inspection is taking place, (b) the development of visual display systems that provide enhanced images for examination and interpretation, (c) a better understanding of the physical phenomena employed in NDE, and (d) the development of new NDE techniques. Bernard[5] described the basic principles of time-of-flight ultrasonics, and discussed its use in inspecting a pressure vessel with steel walls that were almost one foot thick.

Muscara[6] has reviewed the current status and future plans for the USNRC research program on NDE. That report describes projects that are developing and evaluating methods both for in-service inspection (ISI) of key reactor components using ultrasonics and eddy-currents and for continuous monitoring of key reactor components using acoustic emissions. Each of the following subsections discusses in more detail the emerging NDE methods that could be applied to key reactor components. Relevant methods being developed both by the nuclear power industry and by other industries, such as fossil power, petrochemical, and aerospace, are included.

14.1.1 Advanced Processing of Data. Ultrasonic inspection is used for the reactor pressure vessel (RPV), piping, and nozzles. The synthetic aperture focusing technique (SAFT) for ultrasonic testing (UT) has been developed to provide enhanced visual images of flaws detected in components during inspection.[6] The authors of Reference 7 reviewed their field experience with the application of SAFT-UT to the inspection of welds in the recirculation piping system at the Dresden-3 and the Vermont Yankee boiling water reactors (BWRs). The authors' initial results are encouraging, but they indicated that destructive examination of samples from the sites of the flaw indications are required to fully qualify the results of this new technique.

Figure 14.2 (from Reference 7) shows a conceptual diagram of a SAFT field system. It is evident that the real-time processing involved requires a major commitment of computer resources. However, with the rapid advances being made in digital computer technology and parallel processing, such resources will become readily available at reasonable costs within the next few years.

Bernard[5] described a case history where SAFT-UT was successfully used to inspect a refinery's internally stainless-steel-clad, ferritic-steel pressure vessel. The vessel was inspected through the thickness of its ∿12-in. wall. SAFT was used to distinguish between indications from reflectors and flaws.

The authors in Reference 8 reviewed emerging computer-aided UT inspection systems for the automated examination of nuclear piping. That review provides a detailed summary of the hardware and software that is used in six of those pipe-inspection systems. The systems can be controlled automatically and remotely, and the hardware is assembled in reasonably small packages. An adaptable track-mounted, automated pipe scanner is used with four of the systems. All of the systems provide fairly extensive data recording. The computerized ultrasonic data acquisition and processing system and the ultrasonic data recording and processing system also provide data analysis capabilities. As these systems evolve, it is expected that expert software systems will be incorporated into the data analysis to help the inspector in real-time interpretation of the UT information.

The authors in Reference 9 developed a prototype system for detecting surface cracks in piping using the dc-potential-drop method and have applied it to detecting surface cracks on the inside of a 12-in. diameter austenitic stainless steel pipe. Their equipment includes scanners for both the inside and outside surfaces and a personal computer. They performed a large number of solutions for the electric potential distributions around

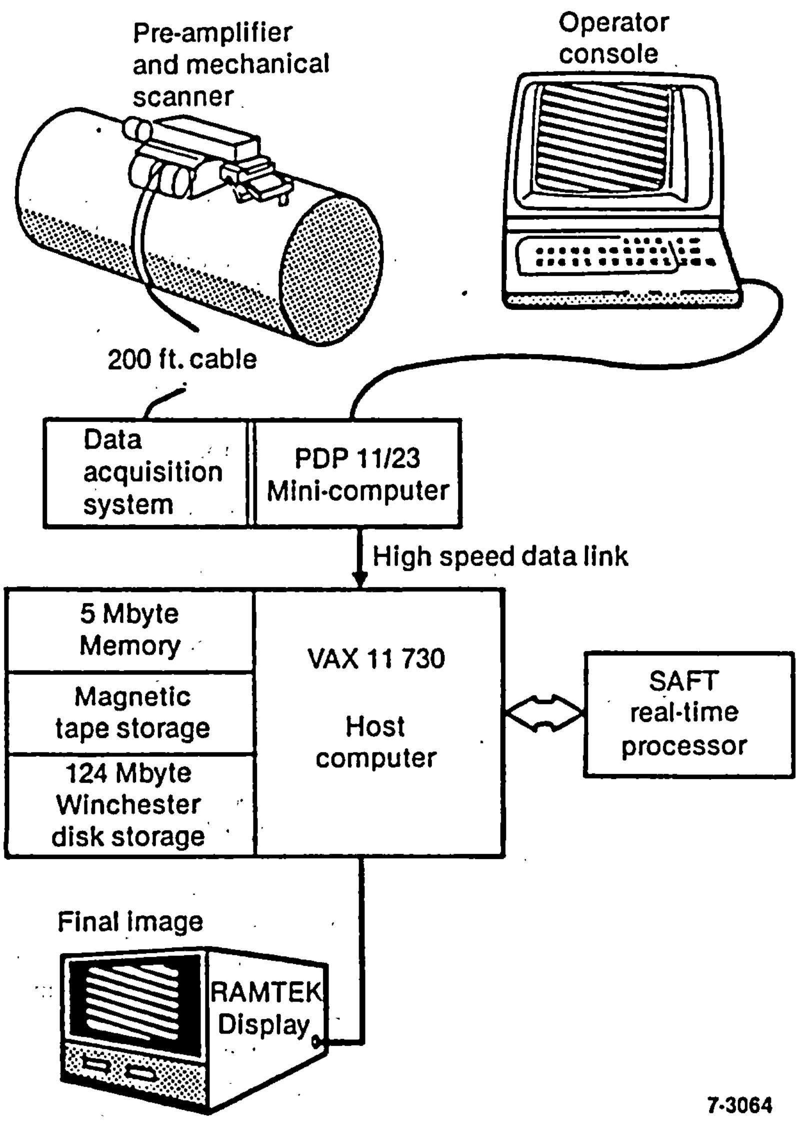


Figure 14.2. Conceptual diagram of SAFT field system.

various sizes and shapes of cracks using the finite-element method. During an inspection, they measured a distribution and compared it with their library of solutions to estimate the crack configuration using the personal computer.

Pressurized water reactor (PWR) steam generator tubes are normally inspected by means of eddy-current testing.[6] The state-of-the-art eddy-current techniques are being evaluated and statistically based reliability data are being developed in an extensive evaluation of a steam generator that has been retired from service.[6] To overcome problems associated with detecting circumferential cracking and intergranular attack, a 16-element pancake-coil eddy-current probe has been constructed and is being evaluated in conjunction with that work.

Whaley and Flora[10] discussed the application of microcomputers to eddy-current testing. They have developed a computerized system for detecting flaws in steam generator tubes near the support plates. They also have performed a preliminary evaluation of the use of a deep-penetration, eddy-current method for detecting flaws in the welds of austenitic stainless steel pipe, with encouraging initial results. However, the computer-aided techniques for eddy-current testing are not as well developed as those for UT.

Thome[11] discussed the application of acoustic (or ultrasonic) holography to defect imaging. To date, this technique has been limited to laboratory evaluation, where it provides excellent images of defects. However, significant development of angle-beam methods is required before this method can be employed to inspect components.

14.1.2 On-Line Monitoring. The USNRC research plan[6] includes two major programs that deal with on-line monitoring of reactor components using acoustic emission (AE) technology. One program is aimed at using AE to monitor crack growth in pressure boundaries,[12] and the other program is aimed at using AE for leak surveillance in light water reactor (LWR) systems.[13]

Methods for AE monitoring of crack growth have been developed in laboratory studies, evaluated during a long-term fatigue test of a pressure vessel, and demonstrated by field surveillance on the Watts Bar-1 reactor.[12] That work has shown that AE sensors and instrumentation can be deployed successfully in a reactor environment. Fatigue-crack growth in pressure vessels and, in some cases, stress-corrosion crack growth in stainless steel piping can be detected by AE monitoring, even with interference from background reactor flow noise. Thus, AE monitoring appears to be a promising technique for on-line monitoring of critical areas in key reactor components.

Kupperman[13] reviewed the current leak detection practices for the cooling systems at 74 nuclear plants. He concludes that the current methods may not be sensitive and reliable enough for detecting some types of possible leaks, especially with regard to the Duane Arnold safe end cracking incident. Work is underway to improve the existing leak-detection systems and to develop two new types of systems—moisture-sensitive tape and AE monitors. Moisture-sensitive tape has been installed in a few reactors for evaluation; it has a tendency to give false alarms and does not provide quantitative information about the leak rate. AE leak detection is being evaluated both through laboratory testing and field demonstration. The initial results are encouraging, but more software development and field validation of the technique are needed before AE leak detection can be used with confidence.

Methods for on-line monitoring of pressure boundary components also are being developed for fossil plants. Davidson[14] describes a prototype system for on-line monitoring of boilers to record and accumulate data during typical operating conditions, during startup, shutdown, and during operating transients. That system is being evaluated at Consolidated Edison's Ravenswood-30 Unit. Similar systems could be installed in a nuclear plant to obtain important and valuable information on key parameters during typical operating histories.

Fry[15,16] discusses the use of neutron-noise analysis for monitoring the conditions in a reactor core. Ex-core neutron noise measurements have been made at 13 LWRs, and a library of data for future reference has been established. This approach shows promise for monitoring vibrations in PWR internals and possibly inadequate core cooling in PWRs. Neutron noise could be used to detect abnormal vibration of instrument tubes and fuel boxes in BWRs, but other vibrations in BWRs were difficult to detect because of the large background noise from boiling. This technique is not yet appropriate for on-line use, but work is continuing to develop it further.

Galpin[17] has outlined a practical, economic approach to either manual or automatic monitoring of component performance to develop information on aging trends. He emphasizes that standard procedures should be employed to monitor parameters, such as temperature, flow, pressure, vibration, and acoustic emission. Repeatable, accurate, and reliable data are needed to document aging trends and use those trends to establish safe operating limits.

14.1.3 Nondestructive Measurement of Materials Properties. Most of the current and emerging NDE techniques are limited to the detection and characterization of flaws and cracks that are in the form of mechanical or elastic discontinuities in components. Recently there has been an increased interest in the nondestructive measurement of materials properties (NDMMP), such as tensile strength, fracture toughness, impact toughness, creep strength, grain size, and precipitate morphology, that heretofore have traditionally been measured by destructive testing and examination of specimens. As illustrated by the papers in a recent volume,[18] the techniques being evaluated for NDMMP include ultrasonic testing, eddy-current testing, and x-ray diffraction analysis. Most of the NDMMP work performed to date has been restricted to laboratory studies. Field applications will depend upon future development of apparatus and techniques.

Generazio[19] investigated the use of ultrasonic attenuation for measuring the mean grain size of copper and nickel. Based on the results of his study, he has suggested a nondestructive method for checking the heat treatment of metals. In cases where aging causes microstructural changes to occur during service at LWR operating temperatures, this type of technique might be used to measure those changes. For example, in studies of gas-turbine alloys, it was found that ultrasonic attenuation could be used to detect and quantify microvoid creep damage.[a] It may be possible to develop similar correlations for radiation-induced damage in key reactor component materials.

Swanson[20] and Vary[21] studied the use of ultrasonic attenuation measurements for ranking material fracture toughness. They have found correlations between ultrasonic measurements and the plane strain fracture toughness of various grades of steel and certain titanium alloys. However, the measurements made by the ultrasonic methods used in these studies required samples with smooth plane parallel surfaces and a very accurate measurement of the sample thickness. Other methods, such as those employing backscattering,[22,23] are being developed to eliminate this restriction, but they require spatial- or directional-averaging techniques to eliminate the effects on the data produced by the coherent nature of grain scattering. The principal drawback of this approach is the complex scanning requirements necessary for spatial and directional averaging. Digital processing methods based on frequency averaging may eventually provide an alternative to the complex scanning requirements of spatial or directional averaging necessary for eliminating the coherent grain scattering effects.

Researchers[24] have successfully made ultrasonic measurements of stress intensity factors caused by surface cracks, but the study was limited to brittle nonmetallic materials. The method was based on the measurement of the reflection coefficient of a Rayleigh wave incident to the crack. Clark[25] extended the use of ultrasonic phase velocity measurements to the measurement of stress and stress intensity factors around crack tips in aluminum. In that work, the ultrasonic method used the birefringence of a horizontally polarized shear wave that occurs in a slightly orthotropic plate in a state of plane stress.

Bussiere[26] reviewed the techniques used for NDMMP, with emphasis on electromagnetic and ultrasonic methods for on-line measurement of the properties of steel. He indicated that eight techniques are useful for NDMMP:

- Magnetic, including Barkhausen noise—for hardness, grain size, precipitate size, morphology, and tensile strength
- Ultrasonic—for formability, grain size, tensile strength, and fracture toughness
- X-rays—for residual stress and stress relaxation
- Resistivity/conductivity—for chemical composition, phase transformations, and state of cold work
- Thermal and thermal imaging—for grain structure and extent of hardening
- Small-angle neutron scattering—for particle, precipitate, and microvoid characterization
- Mossbauer spectroscopy—for microstructure and phase characterization
- Position annihilation—for damage caused by plastic deformation, fatigue, and creep.

None of these techniques have been widely applied to characterize material-aging processes for LWR components, but with further development and the establishment of reference standards some of them may prove useful for that purpose in the future. For example, it may be possible to use magnetic techniques to measure the volume fraction of ferrite and mean ferrite spacing (Chopra and Chung[27] have shown that both of these parameters are important) in stainless steel castings to help assess their potential for embrittlement by thermal aging during long-term service.

Yavelak[28] used Barkhausen noise analysis (BNA) to characterize the service-induced damage in a secondary superheater outlet header that was removed from a 240-MW boiler after 24 years of service. The results indicate that this method has potential for application as a tool to nondestructively measure damage. BNA has been used to inspect and locate damaged regions in headers and steam lines in the field during outages.[29]

Eddy-current methods for measuring material properties are being developed by Clark and Metala.[30] They have used eddy-current techniques to measure applied and residual stress, tensile strength, the amount of temper embrittlement, and the degree of creep damage. Figure 14.3 shows their measured correlations between the yield strength and eddy-current response for a low-alloy steel heat-treated to different strength levels. They point out that a sufficient catalog of standard background data needs to be developed before the eddy-current method can be employed as tool for residual-life assessment.

a. Unpublished results of studies by C. M. Jackson at Battelle Columbus, Columbus, Ohio, 1982.

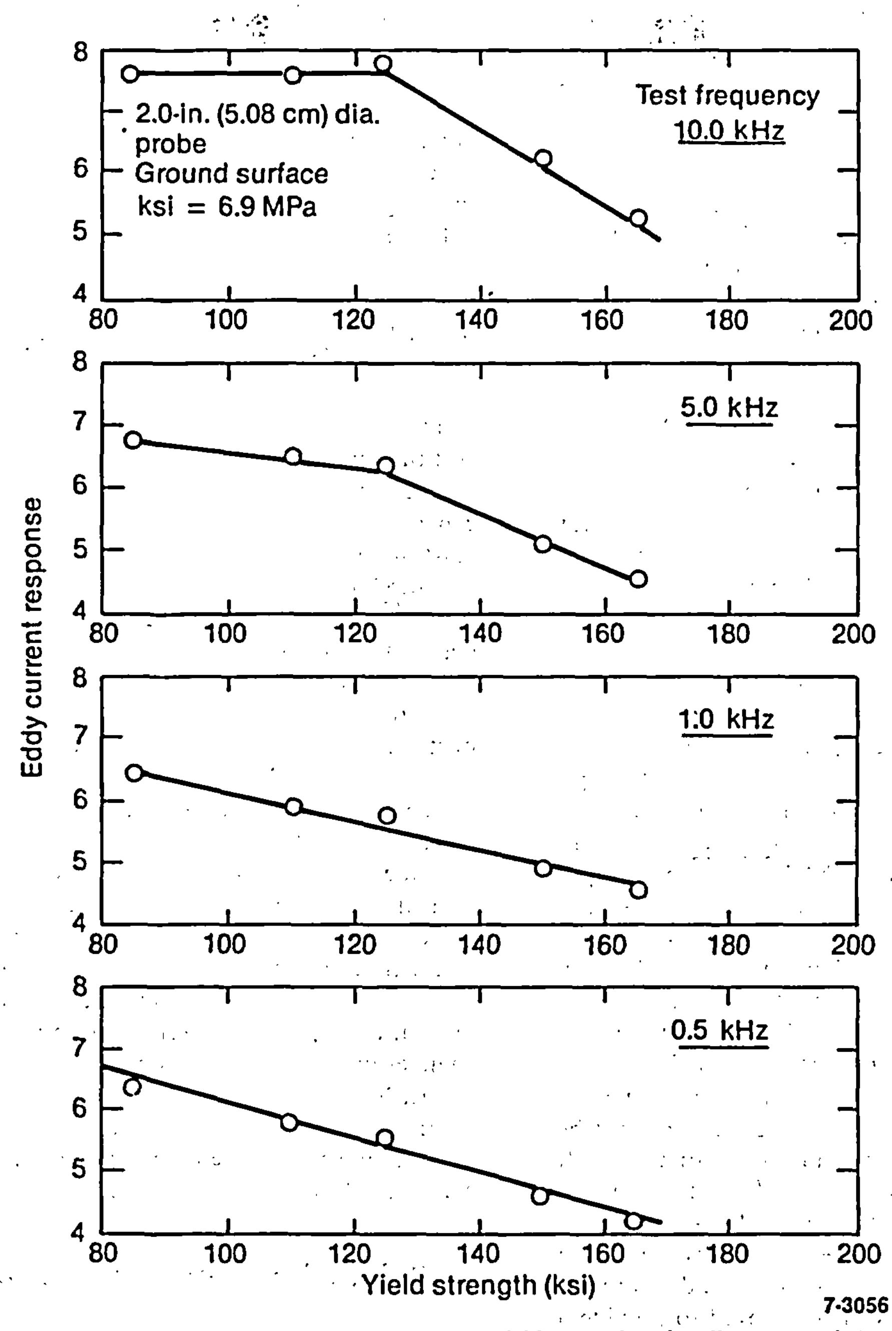


Figure 14.3. Eddy-current response versus yield strength (2-in.-diameter probe).

Electrochemical methods can be used to nondestructively measure material properties. Because of the problem of intergranular stress-corrosion cracking (IGSCC) in BWR piping, an extensive effort was directed toward developing a nondestructive electrochemical testing procedure for measuring the susceptibility of austenitic stainless steels to IGSCC. This work resulted in the development of the electrochemical potentiokinetic reactivation (EPR) test, which has been described and evaluated by Majidi and Streicher.[31] In most cases, the EPR test gives good, quantitative measures of the degree of sensitization with a variety of operators and instruments, and portable equipment is available for performing it in the field. A double-loop EPR test procedure has been developed to simplify the making of field measurements.[32]

Shoji and Takahashi[33] used an electrochemical method similar to the EPR test to evaluate the shift in Charpy transition temperature and the creep damage in steam-turbine rotor and casing steels that result from material degradation during long-term service. They found a correlation between the measured electrochemical parameters and the shift in Charpy transition temperature caused by temper embrittlement and suggested that their method provides a useful tool for nondestructively measuring the degree of temper embrittlement. It may be possible that a similar approach could be employed to nondestructively evaluate the degree of radiation embrittlement in RPV steels.

In heavy-section components, indentation hardness measurements may provide a nondestructive evaluation of the degree of material aging. It may be possible to correlate hardness readings with the degree of radiation embrittlement in RPV steels[34] and then make hardness measurements to assess radiation embrittlement during extended service. As an example, Cane[35] outlined a procedure for using hardness measurements to assess the remaining life of thick-section steel components in fossil power plants.

14.1.4 Surface Replication. Making cellulose acetate replicas of sample surfaces for detailed examination at magnifications up to about 10,000X using electron microscopy has been a typical laboratory procedure for many years. Recently, there has been a great deal of interest in using replication as a tool for field NDE[2,36-43] of components as part of remaining life-assessment activities. That work has concentrated on headers and steam lines in fossil plants rather than on nuclear plant components, but the same techniques could be adapted for use in LWR applications.

Figure 14.4 shows an illustration of the replication and extraction methods from the paper of Masuyama.[2] The surface to be examined must be accessible. The area to be replicated is first cleaned, metallurgically polished, etched, and moistened with an organic solvent. The cellulose-acetate tape (plastic film) is then applied and allowed to sit in place for a time while it softens and adapts to the surface. The replica is then removed to provide a negative image of the surface. For extraction, the etching is deeper so the particles in the microstructure can be removed for examination. To produce high-resolution images, the replicas are coated with carbon or gold in the laboratory and examined using optical or electron microscopy.

GPU Nuclear Corporation has developed a simplified replication technique where mechanical polishing time is minimized by the use of a final electropolishing step and the replica is shadowed with a felt-tip marking pen while it is drying.[42] This approach permits replicas to be made more rapidly (in $\sim$15 min) at the sacrifice of high resolution. These replicas are examined optically at low magnifications using a portable microscope to obtain preliminary information in the field and at magnifications up to 1,000X in the laboratory. A potential danger of such a high-speed technique (electropolishing) is that it may produce artifacts that might be incorrectly interpreted as damage.

For many nuclear-plant applications, a remote replication apparatus would be required. A prototype remote replication unit has been developed for the metallographic examination of steam-turbine rotor bores,[43] and a more advanced rotor-bore remote replication system is planned for development as part of a future Electric Power Research Institute project.[a] Also, a replication manipulator was developed for the remote examination of steam generator tube bores.[44] That equipment was successfully used to inspect regions down to 6 feet in the 0.57-in.-inner-diameter steam generator tubes.[45] In another study, replication was used to detect and document creep-fatigue cracking near welds in a test pressure vessel.[46] The feasibility of using remote replication apparatus has been demonstrated, but specific equipment needs to be developed for use in LWR applications.

Replication may be a valuable inspection tool for areas near welds in LWR vessels, piping systems, and internals. It also may prove useful for inspection of steam generator tubes. It can be used to detect very small cracks or to assess changes in microstructure because of aging. Replicas of cast stainless steels could document the amount, distribution, and spacing of ferrite and might indicate changes in the microstructure related to long-term embrittlement of the ferrite phase. For these and other possible applications, development and demonstration work will be required before replication can be used as a routine ISI tool.

14.2 Life-Assessment Methods

As reviewed previously, the current life-assessment techniques are both experimental and analytical. The newer or emerging methods for life assessment employ miniature test samples that can be removed from a component with negligible damage to the component, on-line procedures for the calculation of damage, and improved models of material degradation and damage accumulation,

a. The project, Steam Turbine Rotor Life Assessment and Extension, EPRI RP2481-5, is scheduled to begin in 1987.

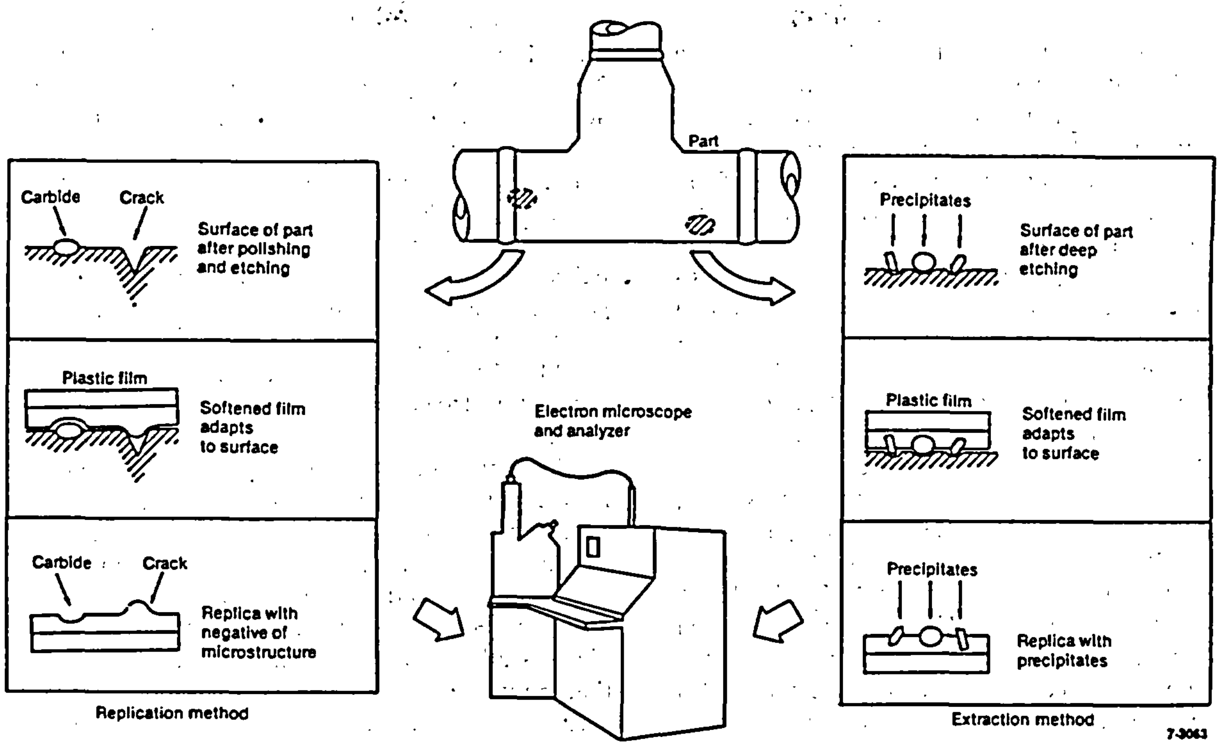


Figure 14.4. Illustration of replication and extraction methods.

especially with the inclusion of probabilistic parameters that are needed to assess safety on a realistic basis. Each of these topics is addressed in the remainder of this section of the report.

14.2.1 Miniature Specimen Testing. Testing of specimens removed from an actual component during service, as mentioned earlier, can provide a direct measure of the degree of aging. If miniature samples are used, then the amount of material that must be removed from the component will be small so that little or no repairs will be required. Miniature specimens also can be valuable for surveillance testing where only a limited amount of test material is available and where space available for material irradiation is restricted. For example, it may be possible to make miniature specimens from the untested grip portions of standard size specimens to obtain additional data. Studies are in progress to evaluate the applicability of miniature specimen technology (MST) to residual-life evaluation.

As discussed by Jaske,[47] there are two main constraints to the application of MST:

- The specimen to be tested must be large enough that it represents the material from which it has been removed
- The specimen to be tested must be small enough that its removal from the component being evaluated does not impair the structural integrity of that component.

The first requirement (minimum size) expresses the need for the miniature specimen to exhibit properties that are the same as those of the bulk material. Experience has shown that the requirement can be satisfied when the minimum specimen dimension is five to ten times as large as the largest strength-controlling microstructural feature. For steels used in LWR components, that feature usually is the grain size, which rarely is greater than 0.1 mm. The minimum-size criterion is applicable, however, only to yield strength, and maybe to ultimate strength.

A shear-punch test has shown promising results for measuring stress-strain response.[48] That test uses small disks of the same size as those used for transmission electron microscopy ($\sim$3.0 by 0.25 mm). During testing, the load versus deflection response of the disk is measured. Finite element stress analysis then is used to develop stress-strain curves. For Type 316 stainless steel, this approach gave results comparable to those obtained from conventional tensile tests. Similarly promising results have been obtained using hardness tests.[49]

Fatigue cracking and fracture toughness, very important properties in remaining life evaluation and reliability analyses, are strongly influenced by grosser microstructural features, such as inclusions and banding.[47] Both of these properties also are notch sensitive, so predictions based on data from unnotched specimens can be extremely misleading. It also is well known that fracture toughness is size dependent; the problem is that small specimens provide nonconservative data compared with data from large specimens. For these reasons, the carbon segregation in specimens from an American Society for Testing and Materials (ASTM) A 508 pressure vessel steel limited the minimum specimen size to 5 mm for obtaining successful measurements of plane-strain fracture toughness, JIc, where the criterion for success was comparison of the results with those obtained from larger specimens that satisfied the criteria of ASTM E 813.[44]

Recently, Misawa and Hamaguchi[50] evaluated two types of miniature specimens for measuring the fracture resistance of ferritic steels. They used a 24-mm-diameter, disk-shaped, compact-type specimen with thicknesses from 1.8 to 5.0 mm to measure the tearing modulus. Also, they used small-punch specimens (0.25x10x10 mm) taken from the ends of broken Charpy specimens to measure the ductile-to-brittle transition temperature and obtained a linear correlation with transition temperature measured using Charpy specimens.

The fracture-toughness data obtained using MST show promise. However, one must be careful in selecting a specimen size for the measurement of fracture toughness and fatigue cracking. Several task groups of the ASTM currently are working on guidelines for the application of MST to fracture-toughness testing.

Battelle's Columbus Division is currently undertaking a major in-house study of MST.[51] The objective of that laboratory study is to evaluate the ranges of parameters over which miniature specimens can be successfully employed. Applications being evaluated include characterization of deformation behavior, fracture toughness, and crack-growth resistance.

Miniature specimens also are being evaluated for use in assessing the remaining life of fossil-plant components. Askins[52] described the development and validation of a miniature-specimen testing technique for evaluating the tensile creep and stress-rupture properties of steels. As is illustrated in Figure 14.5, their specimen design employs a 3-mm-diameter by 13-mm-long piece taken from a component in service. Extension pieces are electron-beam welded to each end of that piece to provide material for gripping in the test fixtures. The final specimen has a configuration with a 2-mm-diameter by 10-mm-long gauge section. For accelerated testing at temperatures near 1000°F (600°C), an argon gas environment is used to avoid erroneous results from excessive oxidation of the test specimen. For a lower temperature of 1000°F (538°C), Saxena[53] tested similarly sized creep specimens from a retired 1-1/4Cr-1/2Mo steel header in air and obtained good agreement with comparable data from tests of standard-size specimens, as is shown in Figures 14.6 and 14.7. Thus, miniature tensile creep testing has been demonstrated to be a viable tool for residual-life assessment.

Although validation work would be required, it is expected that the same type of miniature tensile specimen configuration used for creep testing would provide valid results for the measurement of tensile strength and deformation response of samples removed from reactor components. For example, a study by VanEcho and Williams[a] showed that miniature tensile specimens are useful for measuring the through-thickness properties of thin (~12 mm or less) steel plate. Miniature specimens (including the disk configuration discussed earlier) would be particularly useful for characterizing the properties near welds because samples could be taken from the weld metal and heat-affected zone in cases where standard-size specimens would be too large to sample the properties in the desired region.

14.2.2 Reconstituted Specimen Testing. Another procedure developed to fully use the limited amount of irradiated material that usually is available for evaluation is to reconstitute a Charpy V-notch impact specimen from one half of a previously tested specimen.[54,55] Figure 14.8 illustrates the sequence of steps involved in reconstitution.[54] The fracture face is cut off one half of the tested specimen, and two end tabs are stud welded to it. Final machining and notching of the specimen then is completed. Properly reconstituted specimens yield results that agree with those obtained from the original specimens, as is shown for a typical set of data in Figure 14.9.[54] Thus, the use of reconstituted specimens can triple the amount of Charpy data obtained from the same volume of material when only original specimens are used.

a. Unpublished research results, Battelle Columbus, Columbus, Ohio, July 1981.

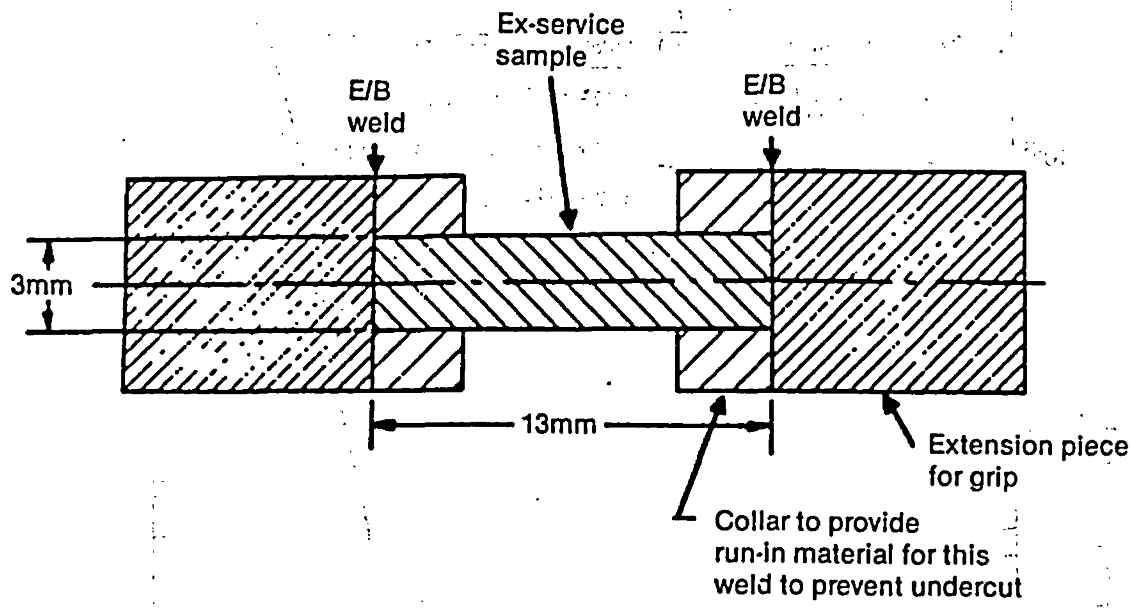


Figure 14.5(a). Schematic diagram of an electron beam welded blank and test piece for tensile testing.

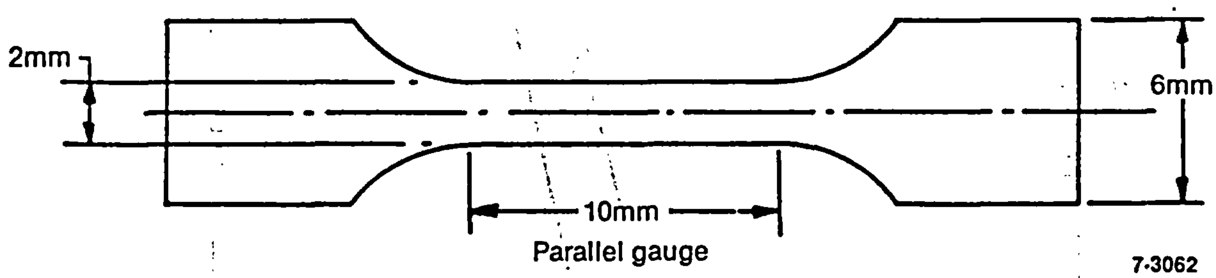


Figure 14.5(b). Schematic diagram of a test specimen machined from the assembly, shown in Figure 14.5(a).

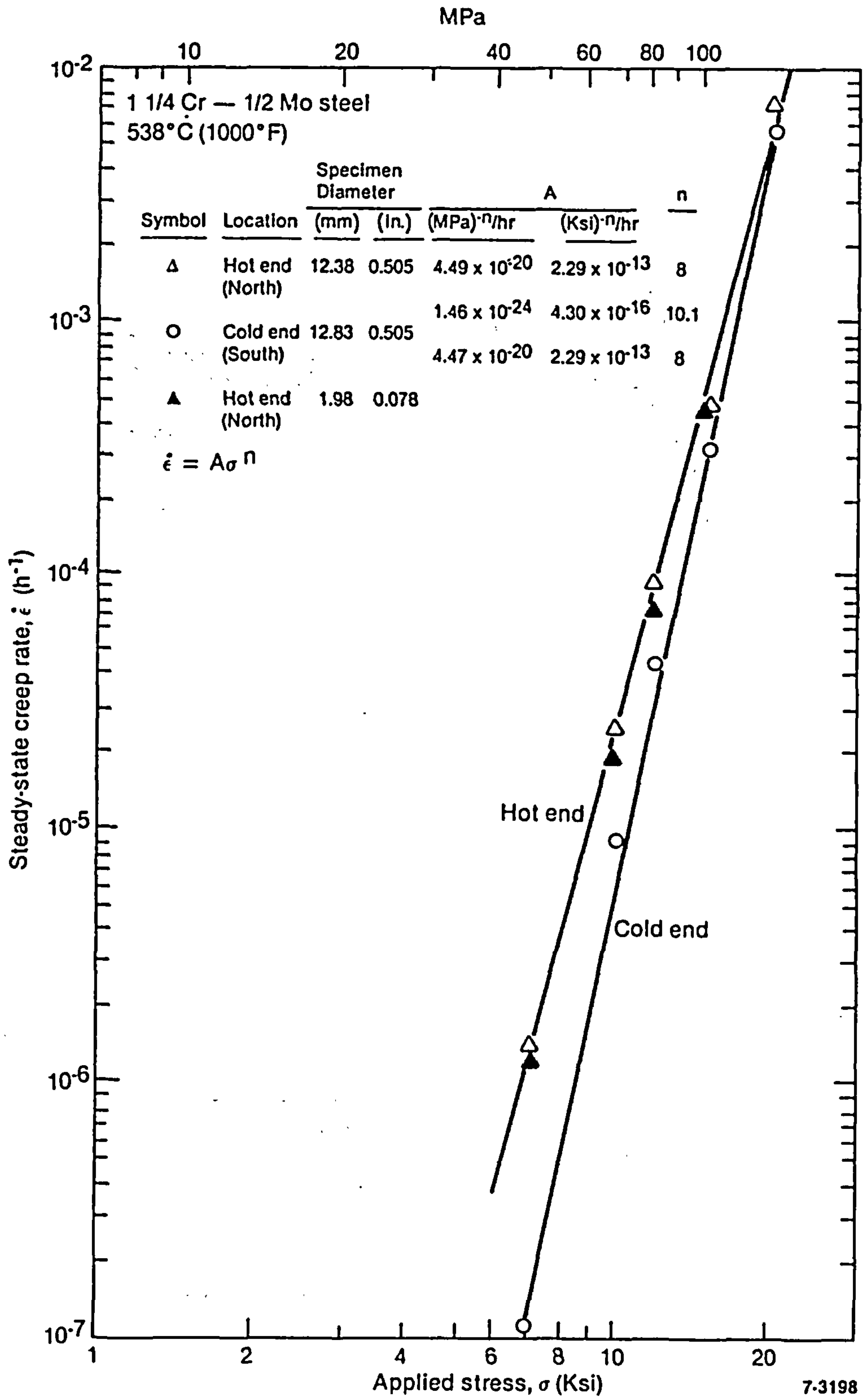


Figure 14.6. Minimum creep rate as a function of stress for hot- and cold-end materials.

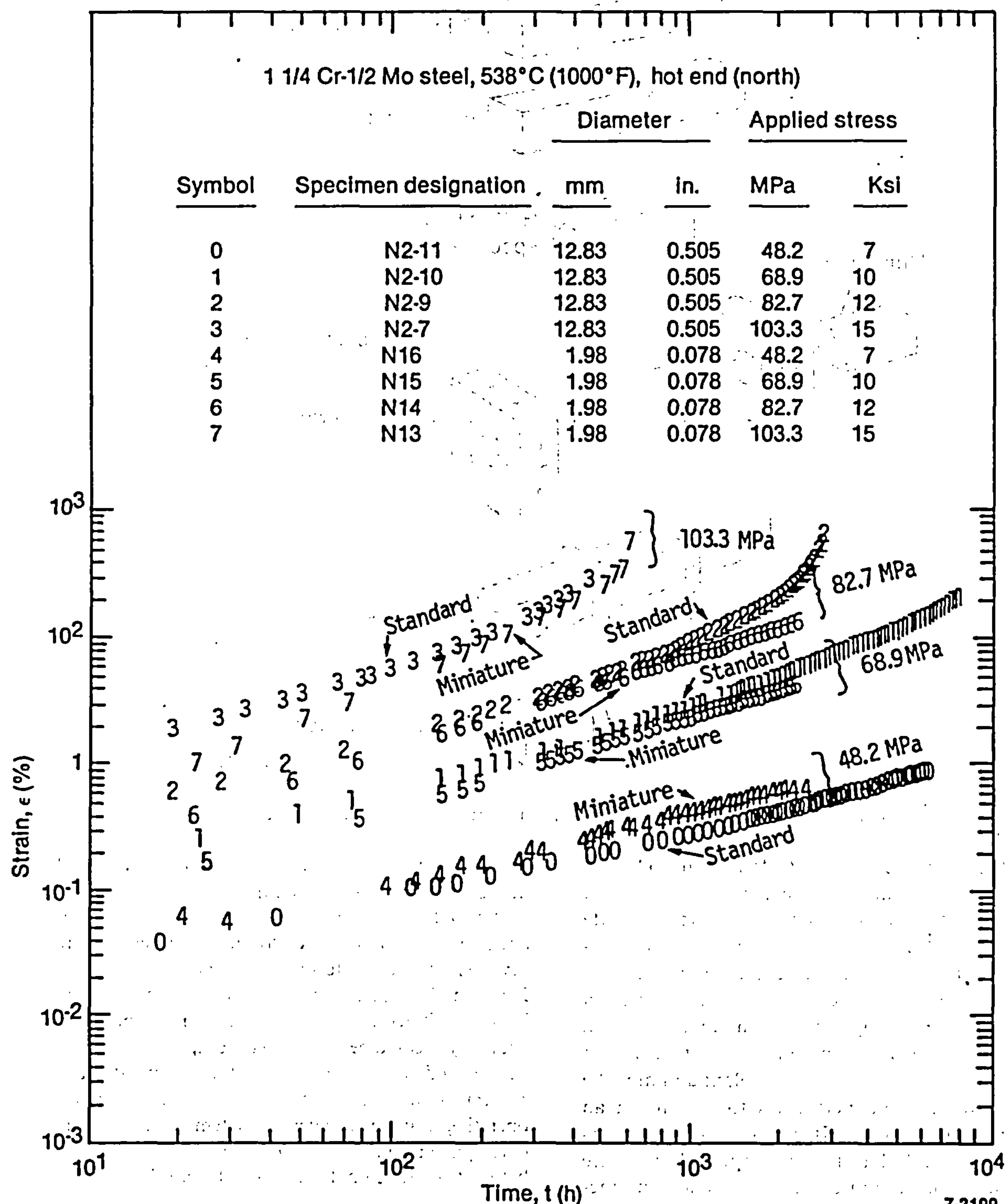


Figure 14.7. Comparison between creep strain versus test time data from standard and miniature specimens.

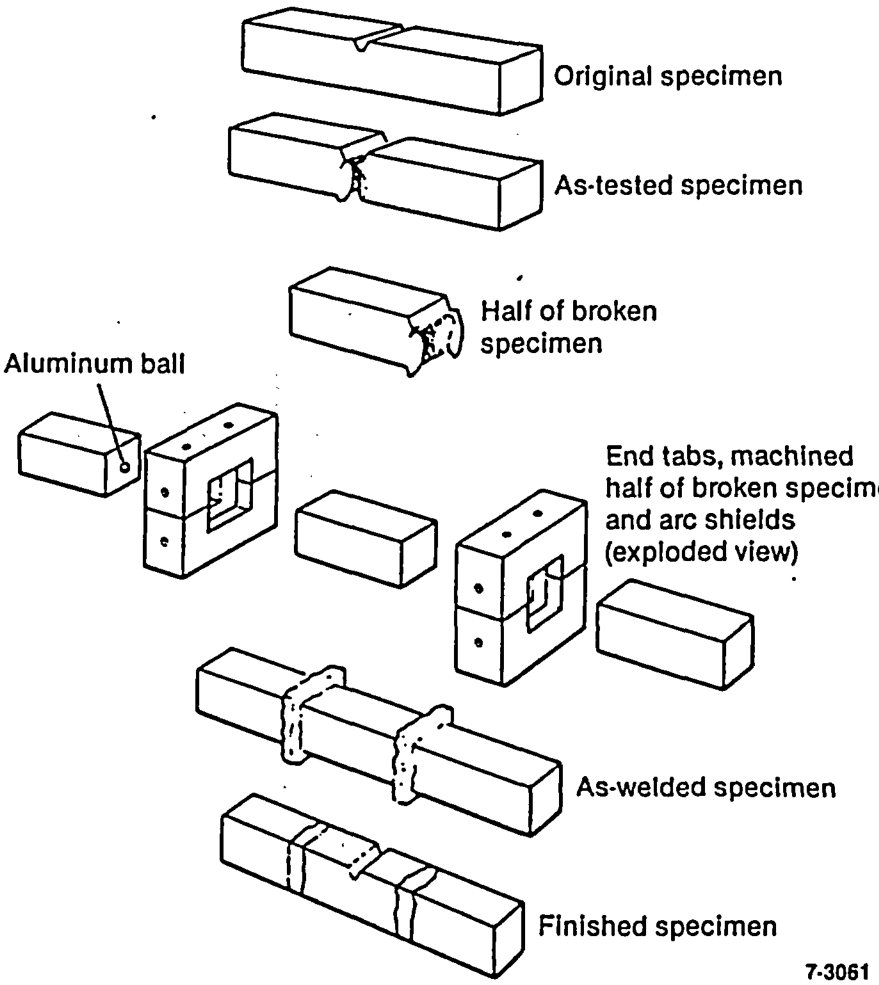


Figure 14.8. Sequence of stages in preparing a reconstituted Charpy specimen.

14.2.3 On-Line Damage Calculation. On-line damage and remaining life calculations are being evaluated as a possible method for residual life assessment.[14] This approach uses a digital computer and thermal, stress, creep, and damage analysis software to compute the damage accumulation during service in conjunction with input data from an on-line temperature monitoring system. All of the equipment is installed at the plant and must be able to survive and perform satisfactorily in an industrial environment. If the calculations are performed in full detail, this approach is quite demanding on computer resources. Thus, in practice, appropriate simplifications must be made to the analyses. If these analyses are performed properly, then the approach appears to be promising and can provide useful information about equipment operation. The major shortcoming, at present, is the lack of validated simple analytical models of the damage accumulation processes.

It would be possible to apply a similar on-line damage accumulation approach to nuclear systems where key operating parameters are being monitored on-line. For example, if local temperatures near a nozzle were being monitored during operation, then on-line damage analysis could be employed to compute the thermal stresses and strains and the resulting fatigue damage accumulation during startups, shutdowns, and major operating transients. Such a system would require development and field demonstration and validation before it could be implemented with confidence.

14.2.4 Effective Engineering Models. Successful implementation of life extension and life assessment strategies requires engineering models that incorporate reasonable simplifying assumptions,

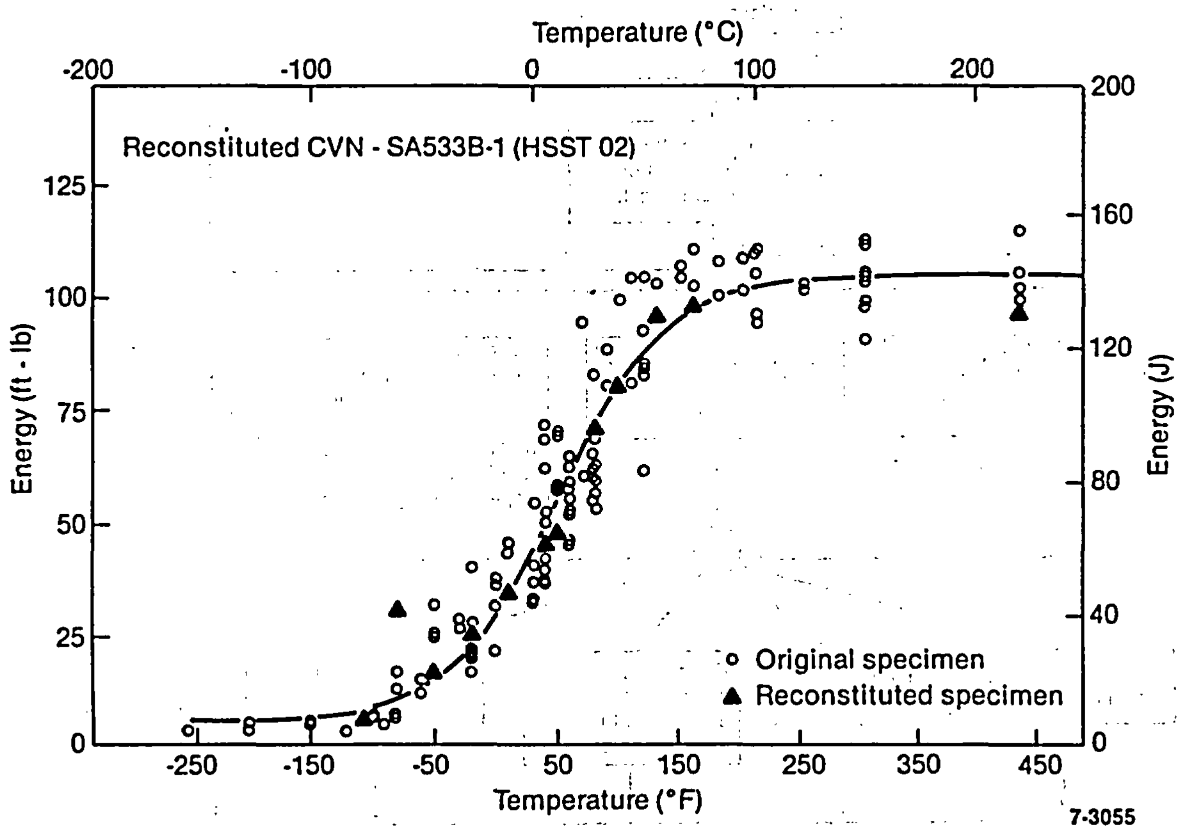


Figure 14.9. Correlation between Charpy data for original and reconstituted specimens.

but still predict the major, important features of material damage evolution (aging). In other words, the models must provide a balance between sophistication and practical utility so that good assessments can be made economically. Where simplifying assumptions are necessary, they should be made on the conservative side to ensure a high probability of safe future operation.

Balkey and Furchi[56] outlined an engineering approach for the life extension of PWR vessels, as is shown in Figure 14.10. First, one makes an initial assessment of the RPV using the current guidelines (Regulatory Guide 1.99). If the required remaining life can be achieved with the current operating parameters, then operations can be continued in the present manner. If not, then fuel management may be evaluated using traditional or new approaches. Other modifications to reduce the neutron flux, such as shielding, may be considered. The appropriate actions are then taken to obtain the desired remaining safe life. This type of assessment is carried out in a probabilistic fashion to help ensure that safe and reliable decisions are made.

Sidey[57] used a model for combined corrosion and stress-rupture damage of fossil-plant superheater and reheater tubes to develop a simplified engineering procedure for estimating the remaining life of reheater tubes. When a reheater tube failed after 78,000 h of service, he used that result to calibrate the model and develop a simple plot of wall thickness versus remaining life, as is shown in Figure 14.11. Now, when ultrasonic testing is used to measure tube-wall thicknesses during an outage, that relationship is used to estimate remaining life of other reheater tubes. Using detailed models to predict remaining life trends based on past operating experience provides a valuable, straightforward engineering approach.

Probabilistic factors need to be incorporated in residual-life estimations. Clark[58] used probabilistic relationships to illustrate the role of NDE in failure risk assessments, as is shown in Figure 14.12. The probability of failure increases as the inspection interval and as the probability of crack initiation, P_i, increase. Increased probability of defect detection, P_d, reduces the probability of failure. This type of approach provides a rational and realistic method for evaluating and minimizing the risk of failure.

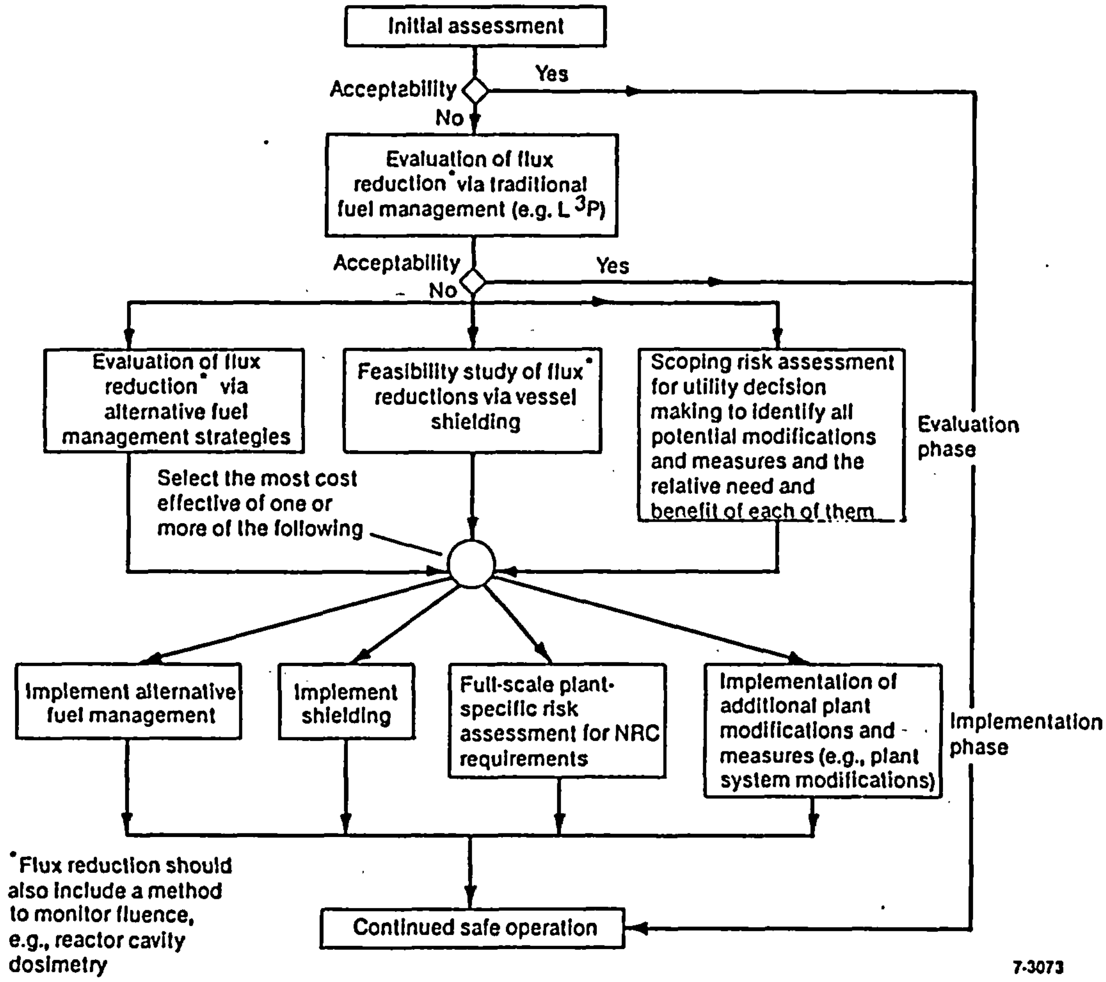


Figure 14.10. A cost-effective approach to address reactor vessel life extension.

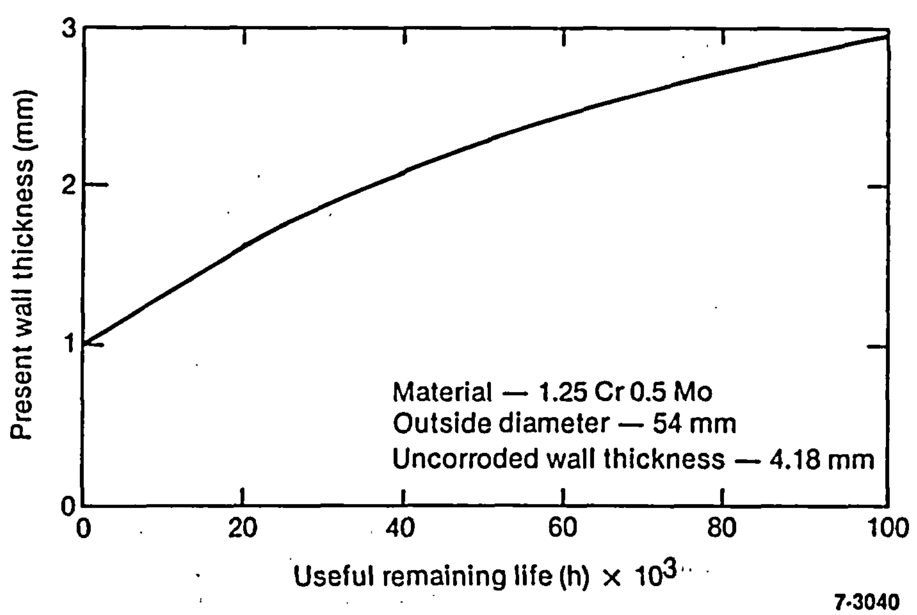


Figure 14.11. Remaining life graph for T11 reheater tubing after 78,000 hours in service.

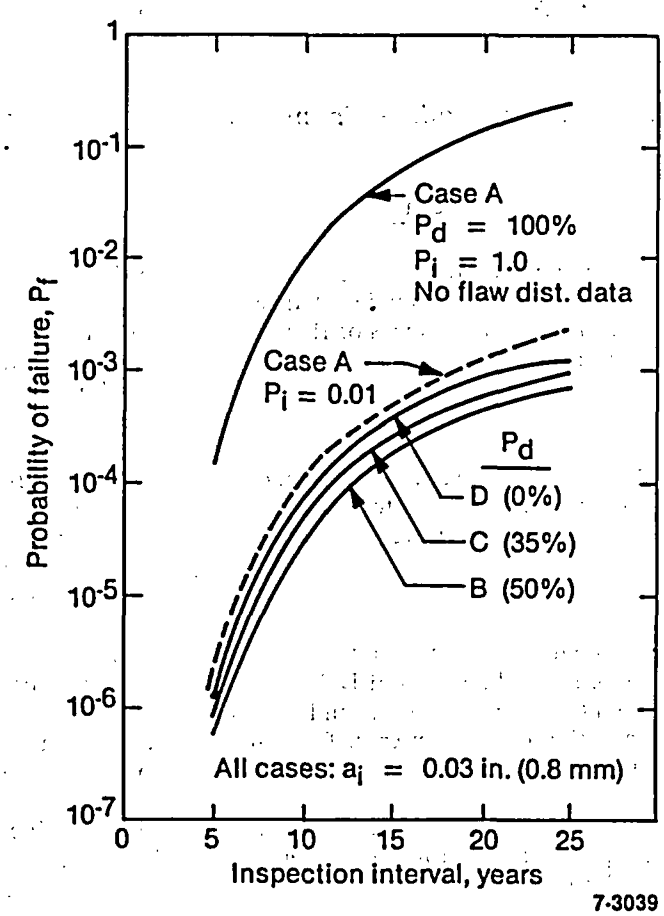


Figure 14.12. Probability of failure versus inspection interval for example cases.

14.3 Summary

The major emerging methods for nondestructive inspection and life assessment of key primary system pressure boundary components are summarized in Table 14.2. For each method, both its characteristics and its applications are briefly highlighted. The applications indicate areas where the method is used and areas where the method has potential use if further advancements and developments are achieved.

Advances in traditional flaw-detection approaches are being achieved through automation and computerized processing of data. With state-of-the-art systems and well-trained, knowledgeable operators, flaws can be located with high probability and sized with moderately good probability. Clearly, there is a need to develop better methods of quantifying the size, nature, and distribution of flaws for use in fracture-mechanics analyses and development work is underway in this area.

Acoustic emission monitoring of crack growth and pressure boundary leaks needs to be further developed for routine field application. On-line neutron-noise and material damage analysis offer long-range potential, but a great deal of development work is required before they can be routinely used with a high degree of confidence.

The nondestructive measurement of materials properties shows a great deal of promise for use in residual-life assessments, but a large amount of research and development is needed before the laboratory techniques can be made commercially viable. Surface replication and miniature specimen testing also offer some potential uses. For real-world use, effective engineering models of material-aging processes are essential. These models must provide a balance between ease of use and complex representations of physical phenomena.

Table 14.2. Summary of emerging methods for inspection and life assessment

Method	Characteristics	Applications
SAFT-UT	Computerized processing of ultrasonic data to provide enhanced significant visual images of flaws in real time.	Welds in the RPV, in piping, and at nozzles. Requires computer resources resources resources and has not been fully developed and verified for field applications.
Computer-aided UT pipe inspection systems	Automated ultrasonic inspection of piping by means of state-of-the-art hardware and software.	Piping and IGSCC cracks near welds in piping. Need more real-time analysis and interpretation of data to aid operators in decision making.
DC potential drop pipe inspection systems	Automated inspection and computerized analysis of DC potential distributions around cracks on both inner and outer surfaces of piping.	Promising new method for inspection of pipes. Prototype system needs to be developed, demonstrated, and verified for routine commercial applications.
Computerized eddy-current inspection systems	Advanced eddy-current coil designs coupled with real-time computer analysis of data.	More reliable inspection of PWR steam generator tubes. The advanced methods need to be fully evaluated and demonstrated. Deep penetration methods show some potential for pipe inspection, but they need considerable development.
Acoustic emission monitoring	Uses computerized processing of data to monitor noise from crack growth or water leakage.	Monitoring of fatigue-crack growth in RPVs, possibly stress-corrosion crack growth in piping, and water leaks in the primary pressure boundary. Requires advanced techniques to remove background noise. Results of initial field studies are encouraging, but further demonstration and validation are needed.
On-line damage analysis	Monitors temperature and pressure during operation and computes damage accumulation on-line in real time.	Creep-fatigue damage in fossil boiler headers and fatigue usage factors in nuclear plant nozzles. Only prototype systems are currently available. Much uncertainty with damage accumulation models that are used.

Table 14.2. (continued)

Method	Characteristics	Applications
Neutron-noise analysis	Measures the ex-core neutron noise and analyzes the recorded data to diagnose potential problems within the RPV.	Shows promise for monitoring vibration of PWR internals and BWR instrument tubes and fuel boxes. Need reference data for normal operations and hardware and software for on-line use.
Nondestructive measurement of materials properties	Uses ultrasonic, eddy-current, x-ray, and electrochemical testing methods to measure properties, such as tensile strength, fracture demonstrated, toughness, and impact strength.	Most applications have been limited to laboratory studies. Offers great potential when methods are developed, and verified by means of future basic and applied research. Need to develop a library of well-characterized correlations between nondestructive measurements and properties.
Surface replication	Produces plastic film replica of material surface that can be examined at high magnifications using both optical and electron microscopy to detect small cracks, voids, and microstructural changes.	Widely used to inspect for creep damage in fossil-plant headers and steam piping. Has potential use for the detection of small cracks and microstructural changes caused by aging in key nuclear components. Could be used to measure ferrite amount and spacing in cast stainless steels. Needs to be developed for nuclear-plant applications.
Miniature specimen testing	Uses a small specimen removed from a component to measure materials properties.	Laboratory studies are being performed to define the range of use for this technique. Shows promise for the measurement of deformation, such as creep and stress-strain response. May have applications for measurement of fracture toughness and crack growth. Further verification of the technique is required for field application.
Effective engineering models	Use of simplified engineering models that provide reasonable descriptions of material aging as a function of operating history.	Trending the aging of key components from a knowledge of the operating history. Most current approaches seem either too detailed and cumbersome to use or too simplistic to provide realistic predictions.

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15. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

The first step in the residual life assessment task was to identify the major light water reactor (LWR) components that are critical to nuclear plant safety during normal operation, off-normal situations, design-basis accidents, or severe accidents. This report has identified and prioritized these components according to their role in preventing the release of fission products to the public. Table 15.1 presents the list of selected pressurized water reactor (PWR) components for the residual life assessment task and reasons for their ranking. This table also identifies the most likely and other potential degradation sites and aging mechanisms for each component. Similarly, Table 15.2 presents the list of selected boiling water reactor (BWR) components.

The next step in the residual life assessment task was to perform a qualitative analysis of the degradation processes active in each of these components and carefully identify corresponding degradation sites, stressors, degradation mechanisms, and potential failure modes during normal operation and accident conditions. This report presented the results of the analyses for seven major components: four PWR components (vessel, containment, primary coolant piping, and steam generator), two BWR components (vessel and recirculation piping), and supports for both PWR and BWR vessels. The currently employed in-service inspection (ISI), surveillance, monitoring, and life-assessment methods were also evaluated. Similar studies will be completed in 1987 of the degradation processes associated with the remaining major components: four PWR components (reactor coolant pump bodies, pressurizers, control rod drive mechanisms, and reactor internals), four BWR components (containments, recirculation pump bodies, control rod drive mechanisms, and reactor internals), cables and connectors, and emergency diesel generators in PWR and BWR plants.

This review of potential or, in many cases, actual degradation sites, mechanisms, and potential failure modes associated with the major LWR components and structures has raised a number of questions or issues deserving further attention. A summary of the results presented in this report, conclusions based on these results, and the recommendations for further work are grouped by component and listed below (these recommendations are indicative of general needs and not necessarily tasks that might be accomplished at the Idaho National Engineering Laboratory as part of the Nuclear Plant Aging Research program). Finally, a summary of the results, corresponding conclusions, and recommendations related to the current and emerging nondestructive examination (NDE) and life-assessment methods is presented.

15.1 PWR Reactor Pressure Vessels

Table 15.3 summarizes the important degradation sites, stressors, degradation mechanisms, potential failure modes, and current ISI and surveillance methods associated with PWR pressure vessels. Note that the ranking of sites is based on the safety impact of that site and not on the possibility of that site having severe degradation. The major conclusions and recommendations are as follows:

1. Radiation embrittlement is the primary safety concern that may limit vessel life extension. Therefore, continued close monitoring of the pressure vessel material condition is required. The current correlations used to estimate the shifts in RT_{NDT} and upper-shelf-energy changes, are uncertain because of the limited accuracy and range of the surveillance data base. The data base does not include the effects of high-fluence and low-flux conditions, and treats the effects of spectrum and irradiation temperature in an approximate manner. A comprehensive plan to obtain reactor pressure vessel (RPV) materials data from environments (i.e. irradiation, thermal, and chemical) that reasonably represent the realities of long-term in-service behavior should be developed.
2. Use of the temperature shift at 30 ft-lb (41 J) level, ΔRT_{NDT}, from the Charpy V-notch curve for certain RPV materials may underpredict the actual temperature shift at the measured transition-temperature fracture toughness of 100 ksi$\sqrt{in.}$ Therefore, the validity of using Charpy test results for estimating temperature shifts to determine transition-temperature fracture toughness needs further assessment. However, the conservatisms in the overall reference toughness curves and in the shifting of these curves may outweigh this difference between Charpy V-notch and fracture toughness tests.

Table 15.1. Key PWR components for residual life assessment

Rank	Component	Reasons for Ranking	Degradation Sites (most likely, others)	Degradation Mechanisms (most likely, others)
1	Reactor pressure vessel	Severe safety impact of catastrophic failure of an embrittled vessel	Weldments in the belt line region, hot-leg and cold-leg nozzles, weldment at the lower shell to bottom head junction, vessel flanges and studs, instrument penetrations in lower head	Neutron embrittlement, corrosion fatigue, thermal fatigue
2	Containment and basemat	Public protection during an accident	Posttensioning system (tendons and anchors), concrete, metal liner, reinforcing steel	Hydrogen embrittlement and stress corrosion cracking of anchor heads, corrosion and relaxation of tendon material, environmental degradation of concrete, corrosion of reinforcing steel and liner
3	Reactor coolant piping and safe ends	Severe safety impact of a large break	Weldments at the safe ends, branch nozzles, cast stainless steel components (elbows and t-connections), complete primary loop piping in new Westinghouse plants	Fatigue, thermal embrittlement
4	Steam generator	Tube rupture will provide a passage from the primary system directly to the environment for the primary coolant, relatively poor operating experience	Inside tube surfaces at U-bends and tube sheet, outside surfaces at tube-to-tube sheet crevices, divider plate, tube support plate, feedwater nozzle, girth weld	Wastage, denting, intergranular stress corrosion cracking, pitting, fretting, intergranular attack, thermal fatigue, corrosion fatigue
5	Reactor coolant pump casing	Primary protection from any internal failure of pump elements (impeller parts, shafts), primary coolant pressure boundary	Casing wall, flanges, seal housing bolts	Corrosion fatigue, thermal embrittlement
6	Pressurizer	Primary coolant pressure retaining boundary	Surge and spray nozzles, spray head, upper shell barrel, seismic lugs	Thermal fatigue, erosion, thermal embrittlement
7	Control rod drive mechanism	Failure may lead to a reactivity initiated accident, an anticipated transient without scram, or a loss-of-coolant accident	Drive rod assembly, control rod pressure vessel	Wear and thermal embrittlement

Table 15.1. (continued)

Rank	Component	Reasons for Ranking	Degradation Sites (most likely, others)	Degradation Mechanisms (most likely, others)
8	Safety-related cables and connectors	Active during normal operation in mitigating operational transients and accidents	Cable insulation, inserts in connectors	Thermal aging and creep of insulation, thermal embrittlement and corrosion of connectors
9	Emergency diesel generator	Needed to operate critical safety equipment in the event of a loss-of-offsite power	Governor in the instrumentation and control system, cooling system pumps and piping, fuel injector pumps, turbocharger	Fatigue and vibrations
10	Reactor internals	Failure may lead to disbursement of fuel into the coolant, or a reactivity-initiated accident	Lower core plate, baffle-former assembly, upper support column bolts, control rod bolts, control rod guide tube sheath and support pins, thermal shield bolts, in-core instrument nozzles, flux thimble tubes bolts, in-core instrument nozzles, flux thimble tubes	Neutron embrittlement, wear, high-cycle fatigue, SCC, relaxation
11	RPV supports	Failure will challenge the integrity of the primary coolant piping and the small lines connected to the RPV; the neutron shield tanks maintain acceptable levels of radioactivity in the containment	Inside surface of the support structure (neutron tank or column support) at the core horizontal midplane level, lubricant in sliding foot assembly of neutron tank support	Neutron embrittlement, corrosion, degradation of lubricant

Table 15.2. Key BWR components for residual life assessment

Rank	Component	Reasons for Ranking	Degradation Sites (most likely, others)	Degradation Mechanisms (most likely, others)
1	Containment and basemat	Public protection during an accident	Concrete, metal liner, and reinforcing steel in suppression pool and primary containment wall, base metal, welds and vent pipes in metal containment	Environmental degradation of concrete, fatigue and corrosion of metal containments
2	Reactor pressure vessel	Primary pressure boundary, poor operating experience with reactor pressure vessel nozzles	Nozzles, closure studs, beltline region, welds at the stub tubes	Thermal fatigue, neutron embrittlement, intergranular stress corrosion cracking
3	Recirculation piping, safe ends, and safety system piping	Primary pressure boundary, relatively poor operating experience	Safe ends, austenitic stainless steel fittings	Intergranular stress corrosion cracking, thermal aging
4	Recirculation pump body	Primary protection from any internal failure of pump elements, primary coolant pressure boundary	Heat affected zones near weldments in the wall	Thermal aging, corrosion fatigue, crevice corrosion
5	Control rod drive mechanism	Failure may lead to a reactivity initiated accident, an anticipated transient without scram (ATWS), or a small-break, loss-of-coolant accident	Drive rod assembly, control rod pressure vessel	Wear and thermal embrittlement
6	Safety-related cables and connectors	Active during normal operation in mitigating operational transients and accidents	Cable insulation, inserts in connectors	Thermal aging of insulation, thermal embrittlement and corrosion of connectors
7	Emergency diesel generator	Needed to operate critical safety equipment in the event of a loss-of-offsite power	Governor in the instrumentation and control system, cooling and lubrication system pumps and piping, fuel injector pumps, turbocharger, generator windings	Fatigue and vibrations
8	Reactor internals	Failure may cause fuel failure or problems in scraming the reactor	Core shroud, top guide plate, core plate, holddown beam in a jet pump, feedwater and core spargers	Irradiation-assisted stress corrosion cracking, intergranular stress corrosion cracking, fatigue
9	Reactor pedestal	Failure will challenge integrity of reactor pressure vessel	Concrete and reinforcing steel	Thermal cycling of concrete, corrosion and fatigue of reinforcing steel
10	Biological shield	Maintains acceptable level of radioactivity in containment	Inside surface of the biological shield at the core horizontal midplane level	Neutron irradiation, gamma-ray heating, environmental degradation of concrete, corrosion of reinforcing steels

Table 15.3. Summary of degradation processes for PWR reactor pressure vessels

Rank of Degradation Site	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Surveillance Methods
1	Beltline region	Neutron irradiation, mechanical and thermal stresses	Irradiation embrittlement (degree is dependent on individual vessel materials and flux spectrum history)	Ductile high-energy tearing leading to leakage (net section over-load)	100% volumetric during first inspection; one weld for subsequent inspection
				Brittle fracture (i.e., pressurized thermal shock)	Surveillance program for assessing irradiation damage is required by law
				Ductile low-energy tearing (low upper-shelf toughness)	
			Environmental fatigue	Ductile overload leading to a leak, possible brittle fracture if PTS occurs	
2	Outlet/inlet nozzles[a]	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload leading to a leak; possible brittle fracture if pressurized thermal shock occurs with some irradiation embrittlement	All nozzle welds inspected volumetrically at each interval
3	Instrumentation nozzles (penetrations) and control rod drive mechanism housing nozzles	Mechanical and thermal stresses	Fatigue crack initiation and propagation	Ductile overload leading to a leak	Visual, inspection of external surface; 25% of nozzles inspected at first interval; remaining 75% spread out over next three intervals
4	Flange closure studs	Mechanical and thermal stresses	Fatigue crack initiation and propagation (possibly corrosion-assisted)	Ductile overload failure (can be replaced)	Volumetric and surface inspection of all studs and threads in flange stud holes at each interval

a. Welds at the vessel to safe end are covered in Chapter 5 on primary system piping; the comments here are also applicable except that volumetric and surface examinations are required.

3. Current RPV materials surveillance programs have limited space and limited numbers of specimens of critical material. Therefore, miniature and reconstituted specimen testing should be performed, so that additional mechanical property data can be extracted from the given limited amount of critical material. Also, the current surveillance programs should be modified as necessary to ensure that specimens are available for surveillance testing during the renewed license period.

4. Thermal fatigue of the RPV nozzles and penetrations is an important safety concern associated with license renewal. Techniques for better monitoring of fatigue transients for nozzles and penetrations should be developed so that the actual usage factors may be determined. To improve the life-prediction capability of the American Society of Mechanical Engineers (ASME) Section III and XI fatigue design analyses, a modified procedure for crack initiation should be developed, based on the local-strain approach that

accounts for the effect of both stress and strain concentrations at notches and of the sequence of variable amplitude cycles. In addition, an updated version of the damage-tolerant procedures that incorporate an improved appreciation of environmental factors, low-growth-rate behavior, and the role of small flaws should be developed.

5. Inspection and transient logging records as well as all material properties (unirradiated and surveillance results), and design data need to be maintained. Collection of these records is needed to develop life-extension plans.
6. Thermal annealing of reactor pressure vessels should be pursued as a long-term option for license renewal. Annealing mechanisms and kinetics need to be better understood and corresponding models need to be developed.

15.2 PWR Containments and Basemats

Tables 15.4, 15.5, and 15.6 (for prestressed concrete, reinforced concrete, and steel PWR containments and basemats, respectively) summarize the important degradation sites, stressors, degradation mechanisms, potential failure modes, and the current ISI, surveillance, and testing methods. The major conclusions and recommendations are as follows:

1. Hydrogen embrittlement, pitting, and microbiological-induced corrosion are the major degradation mechanisms associated with the posttensioning system anchors, tendon wires, and grease in the prestressed concrete containments. Monitoring methods that will detect the degradation of the anchors and decomposition of the tendon grease are needed.
2. Aggressive environments and internal chemical reactions are the major stressors damaging the concrete in concrete containments. Corrosion is the major degradation mechanism associated with the reinforcing bars and steel liners in concrete containments. Additional information about the long-term degradation of reinforced and prestressed concrete containments is needed. Data should be collected from the older LWR containments and facilities that have been shut down after extended service. Accelerated aging techniques could also be investigated and, if found appropriate, used to obtain additional data. The results from these studies should support the development of quantitative residual life models.
3. A comprehensive and standardized ISI program can and should be developed to identify and quantify degradation in reinforced and prestressed concrete containments. State-of-the-art ISI methods used in Europe, Japan, and the United States need to be evaluated and included in the program where appropriate.
4. The impact of the concrete-liner interaction on the integrity of the aged reinforced-concrete and prestressed-concrete containments during severe accidents should be evaluated. The relevant research programs sponsored by the Electric Power Research Institute are expected to provide useful information.

15.3 PWR Coolant Piping and Safe Ends

Table 15.7 summarizes important degradation sites, stressors, degradation mechanisms, potential failure modes, and current ISI methods associated with the PWR coolant piping and safe ends. The major conclusions and recommendations are as follows:

1. Low-cycle fatigue is the main degradation mechanism for the main coolant pipe nozzles and the dissimilar metal welds at terminal ends. However, the number, type, and severity of transients are generally not recorded. Therefore, an in-depth study of past operations in selected nuclear power plants would provide some insight into the rate at which the design life is being consumed because of fatigue.
2. Cast austenitic-ferritic (duplex) stainless steel components, i.e., main coolant pipe, elbows, T-connections, and valve bodies, are subjected to thermal embrittlement. The actual degree of thermal-aging embrittlement of these cast stainless components should be monitored.

Table 15.4. Summary of degradation processes for prestressed concrete containment vessel

Rank	Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes	ISI Method
1	Posttensioning system anchorage	Material properties and trapped water	Hydrogen embrittlement	Loss of stress	Tendon surveillance program
2	Posttensioning tendon wire or strand	Moisture, trapped water, or breakdown of grease material	Pitting, microbiological-induced corrosion	Loss of stress	Tendon surveillance program
3	Steel liner dome and wall	Moisture, acidic environment, and stress	Corrosion and cracking	Liner-concrete interaction, leakage of radioactive gases	Leakage testing (10 CFR 50, Appendix J)
4	Steel liner over base slab	Moisture, acidic environment, and stress	Corrosion	Leakage of radioactive material	Leakage testing (10 CFR 50, Appendix J)
5	Dome, wall, and base slab reinforcing steel	Aggressive environment	Corrosion	Loss of structural integrity	Visual
6	Concrete	Aggressive environment, and internal chemical reactions	Cracking and spalling	Loss of integrity, corrosion of reinforcing steel	Visual, rebound methods, core samples if required

Table 15.5. Summary of degradation processes for reinforced concrete containment vessel

Rank	Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes	ISI Method
1	Dome and wall reinforcing steel	Aggressive environment	Corrosion	Loss of structural integrity	Visual
2	Base slab reinforcing steel	Aggressive environment	Corrosion	Loss of structural integrity	Visual
3	Steel liner over dome and wall	Moisture, acidic environment, and stress	Corrosion	Liner-concrete interaction, leakage of radioactive gases	Leakage testing (10 CFR 50, Appendix J)
4	Steel liner over base slab	Moisture, acidic environment, and stress	Corrosion	Leakage of radioactive material	Leakage testing (10 CFR 50, Appendix J)
5	Dome and wall concrete	Aggressive environment, internal chemical reaction	Cracks and spalling	Loss of integrity, corrosion of reinforcing steel	Visual, rebound methods, core samples if required
6	Base slab concrete	Aggressive environment, internal chemical reactions	Cracks and spalling	Loss of integrity, corrosion of reinforcing steel	Visual, rebound methods, core samples if required

Table 15.6. Summary of degradation processes for steel cylinder/steel sphere containment vessel

Rank	Degradation Site	Stressor	Degradation Mechanisms	Potential Failure Modes	ISI Method
1	Shell welds and base metal	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Loss of structural integrity, leakage of radioactive gases	Visual, leakage testing
2	Interface between shell and concrete slab at base of shell	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Loss of structural integrity, leakage of radioactive gases	Visual, leakage testing
3	Discontinuities in the shell such as hatches and penetrations	Stresses, vibration, cyclic loading, aggressive environment	Corrosion	Leakage of radioactive gases	Leakage testing (10 CFR 50, Appendix J)
4	Steel bottom of shell embedded in concrete	Aggressive environment	Corrosion	Leakage of radioactive material	Leakage testing (10 CFR 50, Appendix J)
5	Base slab concrete	Aggressive environment, internal chemical reactions	Cracks and spalling	Corrosion of reinforcing steel, corrosion of steel bottom of containment shell	Visual, rebound methods, core samples if required

Table 15.7. Summary of degradation processes for PWR primary system piping

Rank of Degradation Site	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI Methods
1	Main coolant pipe nozzles[a]	System operating transients	Fatigue crack initiation and propagation	Through wall leakage	Volumetric inspection for diameter ≥4-in.
2	Terminal end dissimilar metal weld[b]	System operating transients	Fatigue crack initiation and propagation	Through wall leakage	Surface inspection for diameter <4-in.
3	Cast stainless steel	System operating transients	Thermal embrittlement leakage	Through wall leakage	Volumetric

a. The nozzles in the primary coolant piping are the highest ranking degradation sites. Additional specificity is avoided because the most severely loaded nozzles are difficult to determine. Reported results are heavily dependent on type of analysis performed. Actual damage will completely depend on the actual transient usage that occurs in the plant. This is the basis for the recommendation.

b. In Westinghouse plants, the dissimilar metal welds are at the reactor vessel and steam generator nozzles.

15.4 Steam Generators

Table 15.8 summarizes the important degradation sites, stressors, mechanisms, potential failure modes, and current ISI methods associated with the recirculating and once-through steam generators. The major conclusions and recommendations are as follows:

1. Intergranular stress corrosion cracking (IGSCC) and intergranular attack (IGA) are the major degradation mechanisms for the

Table 15.8. Summary of degradation processes for steam generator tubes

Rank[a]	Degradation Site	Stressors	Degradation Mechanisms	Failure Modes	ISI Method
1	Inside surface of U-bends and roll-transition regions	Tube rolling stresses, Corion phenomenon, denting	IGSCC	Cracking	Eddy-current testing
2	Outside surface of hot-leg tubes in the tube-to-tube sheet crevice region	Alkaline environment, presence of SO_4 and CO_3 anions	IGA, IGSCC	May eventually result in cracking	Eddy-current testing
3	Cold-leg side in sludge pile or where scale containing copper deposits is found	Brackish water, air, and copper	Pitting	Local attack and tube thinning may eventually lead to a hole	Eddy-current testing, optical scanner system, sonic leak detector system
4	Outside surface of tubing above tube sheet	Phosphate chemistry, chloride concentration, resin leakage from condensate polisher bed	Wastage (thinning)	Uniform attack, tube thinning may eventually wear out the material	Eddy-current testing
5	Tubes in the tube-support regions	Oxygen, copper oxide, chloride, temperature, pH, crevice conditions	Denting	Flow blockage in tubes caused by plastic deformation	Helium leak and sonic leak testing, optical probes, hydrogen evaluation monitoring, pulse echo ultrasound method
6	Contact between tube and anti-vibration bar	Flow-induced vibrations	Fretting	Wear out of material caused by rubbing and/or fatigue	Eddy-current testing
7[b]	Once-through steam generator tubes	Velocities, sizes, shapes, impact angle, and hardness of particles	Erosion-corrosion from impingement of particles	Wear out of material	Eddy-current testing
8[b]	Once-through steam generator tubes in the upper tube sheet region	Chemicals, (localized corrosion) vibrations	Fatigue	Primary to secondary leaks	Eddy-current testing

a. Based on operating experience for steam generator defects.

b. Denotes once-through steam generator (Items 7 and 8 do not reflect rank order). The first six items are for recirculating steam generators.

recirculating steam generator tubes. Pitting is another important degradation mechanism which progresses at an accelerated rate once initiated, and failures often occur suddenly. Therefore, methods to monitor the degradation of steam generator tubes caused by IGSCC, IGA, and pitting are needed. Also, a better understanding and better models of these corrosion mechanisms are needed.

2. Several changes in the nuclear power plant water chemistry, and the design and materials of construction of the recirculating steam generators (SG) and/or other plant components have been effective in mitigating the degradation damage. The water chemistry changes include use of an all-volatile treatment for the secondary-side water chemistry rather than a phosphate treatment, addition of boric acid to the secondary coolant, use of full-flow condensate polishers, and use of tube sheet crevice flushing. The changes related to the plant design and materials include a change from copper to titanium or stainless steel as condenser tube material, use of stainless steel for the feedwater lines, use of a quatrefoil design and Type 405 stainless steel for the SG tube support plates, SG tube rolling to the full depth of the tube sheet, shot peening of the tubing inside surfaces in the U-bend and roll-transition regions of the SG tubes, and use of thermally treated Inconel 690 in place of Inconel 600 for the SG tubes. The gap size between the antivibration bars and the SG tubes has also been reduced in some designs to avoid fretting. The effects of these mitigating techniques on the failure rates of the SG tubes and tube support plates should be monitored.
3. Erosion-corrosion and environmental fatigue are the major degradation mechanisms associated with the once-through steam generators. The damage caused by environmental fatigue has been reduced by blockage of the untubed lane with a lane blocker.
4. Bobbin-type eddy-current coils have not been successful in all cases in detecting IGA defects. However, successful detection has been achieved by the use of pancake coils that are arrayed with their axes parallel to the tube radius and multiplexed to permit use of available electronic packages.

15.5 Reactor Pressure Vessel Supports

Table 15.9 summarizes the important degradation sites, stressors, degradation mechanisms, potential failure modes, and current ISI methods associated with the five different types of RPV supports: neutron shield tank, column, cantilever, bracket, and skirt-type supports. The first four types of supports are used with PWR RPVs, while the fifth type of support is used on BWR RPVs. The major conclusions and recommendations are as follows:

1. The neutron shield tanks, columns, and cantilever-type supports may be susceptible to catastrophic brittle failure caused by irradiation embrittlement. The neutron spectra have a very large component of low-energy neutrons ($E < 1$ MeV), and the support temperatures are less than 450°F (232°C). Both the fracture toughness and strength of the RPV support steels at high fluence and low irradiation temperatures need to be determined. Also, a correlation of fracture toughness with Charpy V-notch properties at temperatures less than 450°F (232°C), and as a function of displacements per atom is needed.
2. A better determination of the radiation environment (neutron spectra and flux levels) is required for the neutron shield tank, cantilever, and column-type support structures.
3. The dry lubricant in the sliding foot assembly of the neutron tank support degrades because of neutron irradiation. The effects of the radiation levels on the lubricants that are used in the RPV nozzle supports should be investigated.

15.6 BWR Reactor Pressure Vessels

Table 15.10 summarizes the important degradation sites, stressors, degradation mechanisms, potential failure modes, and current ISI and surveillance methods for the BWR pressure vessel. The major conclusions and recommendations are as follows:

1. Low-cycle fatigue of the RPV nozzles is the major degradation mechanism of concern. Refined cycle counting and transient

Table 15.9. Summary of degradation processes for PWR and BWR RPV supports

Rank	Degradation Site[a]	Stressor	Degradation Mechanism	Potential Failure Modes	ISI Method[b]
1	Neutron shield tank at the core horizontal midplane elevation	Neutron irradiation, tensile stresses, operating temperature, water chemistry	Neutron embrittlement, corrosion	Catastrophic brittle failure	Monitoring[c]
2	Column support at the core horizontal midplane elevation	Neutron irradiation, tensile stresses, operating temperature	Neutron embrittlement	Catastrophic brittle failure	Monitoring and sampling
3	Cantilever support in the active height of the core	Neutron irradiation, tensile stresses, operating temperature	Neutron embrittlement	Catastrophic brittle failure	Monitoring and sampling
4	Threaded parts in sliding foot assembly	Tensile stresses, operating temperature	Stress corrosion cracking	Binding that causes possibly excessive stresses in the primary system during heatup and cooldown	—
5	Dry lubricant in sliding foot assembly	Neutron irradiation, operating temperatures	Degradation caused by neutron irradiation	Binding that causes possibly excessive stresses in the primary system during heatup and cooldown	—
6	Skirt support	Mechanical and thermal stresses	Fatigue	Ductile overload failure	—

a. The bracket and skirt RPV supports will experience no neutron embrittlement.

b. There are no national standard ISI methods per se to determine state of degradation.

c. Monitoring of neutron field near RPV support.

severity estimates should be used to monitor nozzle fatigue usage so that actual fatigue usage factors can be determined.

2. Closure studs, flange bushings, and stud holes in flanges without bushing are subjected to wear, fretting, and corrosion. These components should be examined and replaced or repaired as appropriate near the 40-year end of life.
3. Brittle failure of a BWR vessel caused by irradiation embrittlement is extremely unlikely because of the relatively low neutron flux and fluence in the beltline region. However, the effects of irradiation embrittlement of BWR vessels should be monitored. Current RPV materials surveillance programs should be modified for license renewal consideration.
4. Many of the older BWRs have limited accessibility for external ISI of the welds in the beltline region. ISI methods for inspection of these welds from the vessel inside surface need to be developed.

15.7 BWR Recirculation Piping

Table 15.11 summarizes the important degradation sites, stressors, mechanisms, potential failure modes, and current ISI methods for the BWR recirculation piping. The major conclusions and recommendations are as follows:

1. Aging degradation of the recirculation piping caused by IGSCC has been a significant problem at a number of BWRs. The

Table 15.10. Summary of degradation processes for BWR reactor pressure vessel

Rank of Degradation Site	Degradation Site	Stressors	Degradation Mechanisms	Potential Failure Modes	ISI/Surveillance Methods
1	Nozzles (including instrument and CRD penetrations) plus safe end welds	Mechanical and thermal stresses	Fatigue crack initiation and propagation, IGSCC	Ductile overload leading to a leak	All large nozzle welds inspected volumetrically at each interval; visual, external surface inspections of small nozzles/penetrations
2	Closure studs, flange bushings, stud holes	Mechanical and thermal stresses	Fatigue crack initiation and propagation, fretting, corrosion	Ductile overload failure (can be replaced)	Volumetric and surface inspections of all studs, threads in flange stud holes and bushings
3	Beltline region	Irradiation embrittlement	Neutron irradiation (extent depends on vessel materials)	Ductile high-energy tearing leading to a leak (not a serious problem)	100% volumetric inspection; surveillance as required by federal law
4	External attachments	Mechanical and thermal stresses	Fatigue	Ductile overload failure	Volumetric and surface inspections

Table 15.11. Summary of degradation processes for BWR recirculation piping

Rank	Degradation Site	Stressors	Degradation Mechanisms	Failure Modes	ISI Method
1	Weld heat-affected zones furnace sensitized safe ends	Tensile stress, oxygen environment, sensitized heat-affected zones	IGSCC	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape
2	High thermal stress regions predicted by stress rule index analysis	Cyclic tensile stress, corrosive environment	Thermal fatigue, corrosion fatigue	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape
3	Austenitic-ferritic stainless steel castings with high delta ferrite levels	High-temperature, tensile stress, shock loads	Thermal embrittlement	Cracks, leaks	Ultrasonic examination, moisture-sensitive tape, impact test specimens
4	Shaft sleeves, other crevices	Corrosive environment, oxygen-starved areas	Corrosion	Material wastage, leaks	Moisture-sensitive tape

heat-affected zones of the recirculation piping welds have been particularly susceptible. Three conditions are required for IGSCC of austenitic stainless steel to exist: a sensitized microstructure, high tensile stresses, and an oxygenated environment. Several techniques have effectively mitigated IGSCC damage. These techniques include solution heat treatment of the welds, heat sink welding, induction heating stress improvements of the welds, last-pass heat-sink welding, mechanical stress improvement processes, weld overlays, and use of hydrogen water chemistry. Nuclear-grade stainless steels are much less susceptible to sensitization and more resistant to IGSCC, and therefore are used as alternate materials for the recirculation piping material.

2. The possible detrimental effects, if any, of hydrogen additions to the BWR recirculation piping loop need to be evaluated. Addition of hydrogen will reduce the oxygen level in the BWR coolant and may lead to degradation of the feedwater lines because of a erosion-corrosion mechanism similar to that which caused the recent Surry pipe break.
3. A program should be developed to monitor the actual degree of thermal-aging embrittlement in the cast stainless steel components in the recirculation piping loop.
4. The long-term fatigue behavior of the weld overlays needs to be better understood.

15.8 Nondestructive Examination Methods

The NDE methods used in standard practice are visual examinations, penetrant and magnetic particle testing, radiographic examinations, eddy-current testing, ultrasonic testing, and acoustic-emission monitoring. These standard NDE methods, which are employed to satisfy the present ASME code ISI requirements, are not entirely adequate for residual life assessments. Inspection methods are needed to accurately determine the size, shape, location, orientation, and type of both surface and internal flaws, so that fracture mechanics approaches may be used for life assessment. Several emerging methods being developed to satisfy these needs are summarized in Table 15.12. Several emerging inspection methods that can be used to detect time-dependent changes in microstructural features (such as the ferrite phase in duplex stainless steels) are also summarized in Table 15.12. Recommendations for further work on emerging inspection methods are as follows:

1. Development of the synthetic aperture focusing technique (SAFT) should be continued. SAFT ultrasonic testing (UT) can provide enhanced visual images of flaws detected during inspection. The initial results of the application of SAFT-UT in the nuclear industry are encouraging.
2. The dc potential drop method and computer-aided eddy-current testing techniques need to be evaluated and their ability to detect flaws in pipe needs to be determined.
3. The use of acoustic emission technology for on-line monitoring of crack growth in pressure boundaries, and leak surveillance in LWR systems should be evaluated.
4. NDE methods for measuring the material properties changes caused by aging of the major LWR components are needed. Some of the promising methods are eddy-current methods to measure residual stresses, electrochemical methods to measure the susceptibility of austenitic stainless steel to IGSCC, and indentation hardness measurements to assess radiation embrittlement of RPV steel.

15.9 Emerging Life-Assessment Methods

Emerging life-assessment methods include miniature specimen testing (MST), reconstituted specimen testing, on-line damage monitoring, and effective engineering models applications. Table 15.12 summarizes two of these life-assessment methods. These methods may provide valuable information concerning several important degradation mechanisms. For example, MST can be used for surveillance testing where only a limited amount of test material is available. On-line damage monitoring can be applied to calculate fatigue usage factors for RPV nozzles during startups, shutdowns, and major operating transients, where nozzle temperatures are monitored during operation. The engineering models can then be used—along with the information from the improved ISI methods, data from MST activities, and on-line monitoring data—to properly estimate the residual life of various components. These estimates will form the basis for LWR license renewals. Recommendations for further work on the emerging life-assessment methods are as follows:

1. The usefulness for RPV surveillance testing of the miniature specimen and reconstituted specimen testing methods needs to be evaluated. The validity of the tensile property, fracture toughness, and fatigue-crack initiation and growth data obtained with these two methods also needs evaluation.
2. Development, field demonstration, and validation of the on-line damage monitoring methods are required before they can be implemented with confidence.
3. Simplified engineering models should be developed to provide reasonable estimates of material aging as a function of operating history.

Table 15.12. Summary of emerging methods for inspection and life assessment

Method	Characteristics	Applications
SAFT-UT	Computerized processing of ultrasonic data to provide enhanced significant visual images of flaws in real time.	Welds in the RPV, in piping, and at nozzles. Requires computer resources resources resources and has not been fully developed and verified for field applications.
Computer-aided UT pipe inspection systems	Automated ultrasonic inspection of piping by means of state-of-the-art hardware and software.	Piping and IGSCC cracks near welds in piping. Need more real-time analysis and interpretation of data to aid operators in decision making.
DC potential drop pipe inspection systems	Automated inspection and computerized analysis of DC potential distributions around cracks on both inner and outer surfaces of piping.	Promising new method for inspection of pipes. Prototype system needs to be developed, demonstrated, and verified for routine commercial applications.
Computerized eddy-current inspection systems	Advanced eddy-current coil designs coupled with real-time computer analysis of data.	More reliable inspection of PWR steam generator tubes. The advanced methods need to be fully evaluated and demonstrated. Deep penetration methods show some potential for pipe inspection, but they need considerable development.
Acoustic emission monitoring	Uses computerized processing of data to monitor noise from crack growth or water leakage.	Monitoring of fatigue-crack growth in RPVs, possibly stress-corrosion crack growth in piping, and water leaks in the primary pressure boundary. Requires advanced techniques to remove background noise. Results of initial field studies are encouraging, but further demonstration and validation are needed.
On-line damage analysis	Monitors temperature and pressure during operation and computes damage accumulation on-line in real time.	Creep-fatigue damage in fossil boiler headers and fatigue usage factors in nuclear plant nozzles. Only prototype systems are currently available. Much uncertainty with damage accumulation models that are used.

Table 15.12. (continued)

Method	Characteristics	Applications
Neutron-noise analysis	Measures the ex-core neutron noise and analyzes the recorded data to diagnose potential problems within the RPV.	Shows promise for monitoring vibration of PWR internals and BWR instrument tubes and fuel boxes. Need reference data for normal operations and hardware and software for on-line use.
Nondestructive measurement of materials properties	Uses ultrasonic, eddy-current, x-ray, and electrochemical testing methods to measure properties, such as tensile strength, fracture demonstrated, toughness, and impact strength.	Most applications have been limited to laboratory studies. Offers great potential when methods are developed, and verified by means of future basic and applied research. Need to develop a library of well-characterized correlations between nondestructive measurements and properties.
Surface replication	Produces plastic film replica of material surface that can be examined at high magnifications using both optical and electron microscopy to detect small cracks, voids, and microstructural changes.	Widely used to inspect for creep damage in fossil-plant headers and steam piping. Has potential use for the detection of small cracks and microstructural changes caused by aging in key nuclear components. Could be used to measure ferrite amount and spacing in cast stainless steels. Needs to be developed for nuclear-plant applications.
Miniature specimen testing	Uses a small specimen removed from a component to measure materials properties.	Laboratory studies are being performed to define the range of use for this technique. Shows promise for the measurement of deformation, such as creep and stress-strain response. May have applications for measurement of fracture toughness and crack growth. Further verification of the technique is required for field application.
Effective engineering models	Use of simplified engineering models that provide reasonable descriptions of material aging as a function of operating history.	Trending the aging of key components from a knowledge of the operating history. Most current approaches seem either too detailed and cumbersome to use or too simplistic to provide realistic predictions.

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This report presents an assessment of the aging (time-dependent degradation) of selected major light water reactor components and structures. The stressors, possible degradation sites and mechanisms, potential failure modes and currently used nondestructive examinations, in-service inspection (ISI) and life assessment methods are discussed for seven major light water reactor components: pressurized water reactor (PWR) and boiling water reactor (BWR) pressure vessels, PWR containment structures, PWR reactor coolant piping, PWR steam generators, BWR recirculation piping, and reactor pressure vessel supports. Unresolved technical issues related to life extension of these components, including requirements for advanced ISI and life assessment methods, are also discussed.

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