Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)

1008203

Topical Report, June 2005

Non-proprietary Version for NRC
(Proprietary Version was submitted to NRC under Affidavit)

EPRI Project Manager
H. T. Tang

Electric Power Research Institute • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

ATI Consulting

ORDERING INFORMATION

Requests for copies of this report should be directed to EPRI Orders and Conferences, 1355 Willow Way, Suite 278, Concord, CA 94520, (800) 313-3774, press 2 or internally x5379, (925) 609-9169, (925) 609-1310 (fax).

CITATIONS

This report was prepared by

ATI Consulting
6773-C Sierra Court
Dublin, CA 94568

Principal Investigators
R. Nickell
T. Griesbach

This report describes research sponsored by the Electric Power Research Institute.

The report is a corporate document that should be cited in the literature in the following manner:

Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134). EPRI, Palo Alto, CA: 2005. 1008203.

REPORT SUMMARY

This report describes a framework and associated strategies for managing effects of aging in pressurized water reactor (PWR) internals using 1) available knowledge of internals design, materials of construction, and material properties; 2) operating experience and age-related degradation mechanism knowledge that provides susceptibility criteria for known aging mechanisms; and 3) supplementary plant surveillance, testing, monitoring, and examinations, as appropriate for susceptible internals components.

Background
Management of aging effects—such as loss of material, reduction in fracture toughness, or cracking—depends on the demonstrated capability to detect, evaluate, and potentially correct conditions that could affect system, structure, or component function. For PWR internals, utilities have identified the general elements of aging effects management programs, including existing inservice inspection and monitoring, with the possibility of enhancement or augmentation depending on future research and development findings. The aging management framework and strategies in this report are developed based on selective inspections and functionality evaluations, incorporating state-of-the-art design, materials, and degradation mechanisms.

Objectives
To establish a framework and associated strategies for managing aging effects of PWR internals to support development of safe, cost-effective, and technically justified reactor internals inspection and evaluation (I&E) guidelines.

Approach
The principal investigators first summarized degradation mechanisms—such as irradiation-assisted stress corrosion cracking (IASCC), irradiation embrittlement, void swelling, stress relaxation, fatigue, and others—that PWR internals potentially would experience during plant operation. Subsequently, the effects of aging degradation in PWR internals were described. Investigators then developed a PWR internals designs and sample component template and PWR internals component categorization process. Finally, approaches for aging degradation categorization and functionality evaluation of components and approach for aging management of PWR internals using inspections were proposed.

Results
This report provides a framework and an overall strategy for managing effects of age-related degradation mechanisms in PWR internals, with a particular emphasis on managing those

degradation effects during the license renewal period for extended plant operation. The important elements of the framework and strategies are

- screening, categorizing, and ranking internals components for susceptibility to aging degradation mechanisms and
- functionality analyses and safety assessment of internals components to define a safe and cost-effective aging management inservice inspection method and strategy.

EPRI Perspective

This Framework and Strategies Report draws on the existing state of knowledge, available results from the EPRI Boiling Water Reactor Vessel and Internals Project (BWRVIP), utility owners groups, and individual utility license renewal applicants. The report also draws on two previously published EPRI reports: *Materials Reliability Program: Strategies for Management of Aging Effects in PWR Reactor Vessel Internals (MRP-62)* (1006582, February 2002) and *Materials Reliability Program: Strategies for Managing Aging Effects in PWR Reactor Vessel Internals – Interim Update (MRP-99)* (1007848, December 2003). This report is the first of a three-part document developed by the EPRI Materials Reliability Program (MRP) for managing aging effects in PWR internals. The other two parts, which will be published individually in the near future, are on PWR internals aging degradation mechanism screening and threshold values and PWR internals inspection and flaw evaluation approaches.

The EPRI MRP Reactor Internals Issue Task Group (RI-ITG) program is currently focusing on performing a detailed screening and functionality evaluation of the PWR internals components and qualitative safety evaluation of the effects of aging degradation on the potential failure to perform component function. In parallel, hot cell testing to quantify aged/irradiated materials behavior and performance is continuing. This report, together with documents to be issued on ongoing and planned studies, will provide a recommended methodology and related technical basis for utilities planning aging management programs for PWR internals.

Keywords

PWR internals
Aging management
Inspection
Functionality
License renewal

ABSTRACT

Demonstration that the effects of age-related degradation in Pressurized Water Reactor (PWR) internals are adequately managed is important for maintaining a healthy fleet and to assure functionality of the PWR internals components. As part of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Reactor Internals Issue Task Group (RI-ITG), this report proposes an overall framework and associated strategies for managing the effects of aging in PWR internals using 1) available knowledge of internals design, materials of construction, and material properties; 2) operating experience and age-related degradation mechanism knowledge that provides susceptibility criteria for known aging mechanisms; and 3) supplementary plant surveillance, testing, monitoring, and examinations, as appropriate for the susceptible internals components. The framework and associated strategies presented here are based on:

- The categorization of PWR internals components into three groupings - those that can be shown to have a low susceptibility to particular age-related degradation effects (Category A), those that can be shown to be sufficiently susceptible to be identified as "lead" components for supplementary aging management (Category C), and a third grouping of moderate susceptibility components (Category B);

- Functionality and safety evaluations of lead components (Category C), in the degraded condition to determine the actual need for supplementary aging management; and

- Incorporation of supplementary aging management program elements, as appropriate, into one-time or periodic plant surveillance, testing, monitoring, and examination programs to assure structural integrity and functionality of the affected PWR internals components.

The framework and associated strategies incorporates existing and available knowledge of design, materials, and degradation mechanisms from ongoing research programs, including the EPRI MRP RI-ITG, Owners Group programs, License Renewal Topical Reports, NEI/MTAG Degradation Matrix (DM)/Issues Management Table (IMT), and the EPRI BWR Vessel and Internals Program (BWRVIP). Additional data under PWR conditions is expected in a number of areas, including mechanical properties, crack initiation, crack growth, fracture toughness, and void swelling. This information will be used to modify implementation strategies for PWR internals aging management as necessary.

This Framework and Strategies Report is one part of a multiple set of documents developed by the EPRI-MRP for managing aging effects in PWR internals. Related documents will be published on PWR Internals Aging Degradation Mechanism Screening and Threshold Values, and PWR Internals Inspection and Flaw Evaluation Approaches. Other key research results from the EPRI MRP program will focus on a detailed screening and functionality evaluation of the

PWR internals components and qualitative safety evaluation of the effects of aging degradation on the potential failure to perform component function. Together, these documents will provide a recommended methodology and the related technical basis for utilities planning aging management programs for PWR internals.

ACKNOWLEDGMENTS

The authors acknowledge the guidance of the EPRI Project Manager, H.T. Tang, and the input and comments from key contributors during the development of this report: Jeff Gilreath (Duke Energy), Charlie Griffin (Progress Energy), Steve Fyfitch (AREVA), Glenn Gardner (Dominion Energy), and the other members of the Reactor Internals Issues Task Group (RI-ITG) who participated in the review of this document. In particular, Steve Fyfitch's contribution to the re-write of Chapter 2 and the editing of the report is acknowledged.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

The purpose of this report is to provide a framework and strategy for developing guidelines for managing the effects of aging in PWR internals. This report was developed by the EPRI Materials Reliability Program (MRP) Reactor Internals Issue Task Group (RI-ITG). A schematic of the framework is shown in the block diagram of Figure 1-1. The three associated strategies are also shown in Figure 1-1 as action blocks on the fourth tier of the schematic. Those strategies are:

- A categorization strategy for dividing the potentially affected list of PWR internals components into three groups – a group called Category A that consists of components for which it can be shown that the effects of age-related degradation are insignificant, a group called Category C for which the effects are deemed to be sufficiently significant for definition as a lead component, and a third group called Category B that consists of the remaining PWR internals components; the latter group are referred to as components with age-related degradation effects that range from low to moderate; (Note: The use of components throughout this report refers to subcomponents or groups of component items.)
- A functionality and safety evaluation strategy based on analysis of PWR internals components in Category C, and even potentially those in Category B, to determine whether the component can continue to perform its intended functions – including any required safety functions – in the degraded condition [Components that can continue to perform their intended functions in a degraded condition are referred to as damage tolerant (a generalization of the concept of flaw tolerance)]; and
- A strategy for incorporating supplementary elements of surveillance, testing, monitoring, and examination into one-time or periodic aging management programs for those PWR internals components in Category C (lead components), as appropriate.

Figure 1-1 shows that the three strategies are dependent on a number of information blocks, such as the block showing the determination of screening criteria for Irradiation-Assisted Stress Corrosion Cracking (IASCC), crack initiation and growth, void swelling, fracture toughness, and stress relaxation. This information block feeds directly into the "Screen Components" block. Another critical information block is the gathering of plant operating and materials data, which feeds into the "Develop Material Constitutive Equations" block and eventually into the "Screen Components," "Perform Functionality Analysis," and "Identify Lead Component" blocks.

The general framework and the three strategies are somewhat more complex than shown in Figure 1-1, however. For example, the functionality and safety evaluations are capable of re-categorization of components – from Category C to Category B for damage-tolerant components, and from Category B to Category A for damage-tolerant components. The functionality and safety evaluations strategy is also capable of identifying Category C components that are susceptible to age-related degradation effects to an extent that these components should be placed into a Category B'. Components in this category are candidates for an enlarged group of locations subject to supplementary elements of surveillance, testing, monitoring, and examination, should such an enlarged group of candidate locations be needed. This need will be determined by the results from initial industry examinations of the Category C components. In addition, the intent of the strategies is to permit and justify the use of existing aging management program elements for PWR internals components in Category A.

Figure 1-1 also represents a chronological sequence of actions by the EPRI MRP RI-ITG. The information blocks on the second tier, and to some extent on the third tier, refer to other reports that are contemporary with this report, or which follow closely behind this report. The development of constitutive equations is documented in reference [1]. A second report on the screening criteria to be used in the categorization process is documented in reference [2]. A third report on recommended supplementary elements to be considered for aging management programs is documented in reference [3]. All of these steps and strategies are intended to culminate in the eventual action shown as "Issue PWR Internals Inspection Guidelines."

Continued evaluation of the aging degradation effects on functionality will be incorporated as the results become available, and data from research test programs will be examined for improved understanding of the aging mechanisms. A listing of related EPRI reports is given in references [4] through [21]. These ongoing MRP activities will be focused toward developing a final set of guidelines for aging management of PWR internals using evaluations and/or inspections.

This report was structured to follow the strategic logic of Figure 1-1, with Chapter 2 providing discussion of the age-related degradation mechanisms and their effects, together with a set of working criteria for preliminary screening and categorization efforts. Chapter 3 provides information on PWR internals design features and a sample template for organizing vendor materials and component operating parameter information. Chapter 4 describes the component categorization process. Chapter 5 provides several approaches for aging degradation categorization and determining the functionality of components. Chapter 6 provides a description of potential inspection and surveillance program options for the lead components, and Chapter 7 provides a summary.

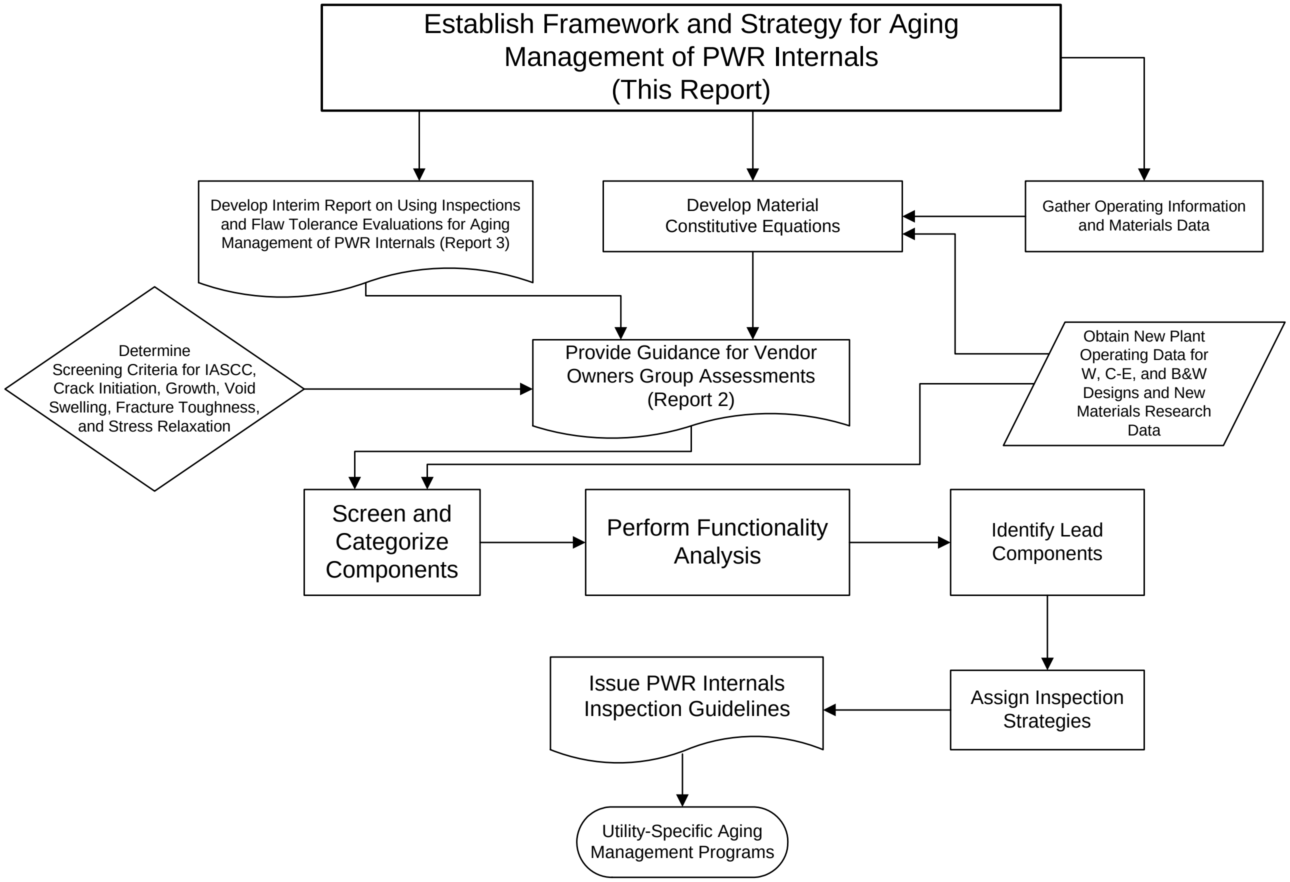

Figure 1-1
Framework for Implementation of Aging Management Using Screening, Functionality Evaluations, and Inspections

2
PWR INTERNALS POTENTIAL AGING MECHANISMS AND EFFECTS

Understanding the mechanisms of aging degradation is important for managing the potential aging effects in PWR internals. PWR internals aging management involves monitoring or predicting the levels of degradation, evaluating mitigation and/or repair techniques, and using inspections or some other type of surveillance to assure component integrity. An age-related degradation mechanism is considered significant if it cannot be shown that the component would maintain its function when the degradation mechanism is allowed to continue without any additional preventive or mitigative measures.

The potential age-related degradation mechanisms for PWR internals materials are discussed in Section 2.1. The relationships between mechanisms and effects are discussed in Section 2.2. A set of screening criteria, which also provides the technical bases behind the criteria, for aging degradation susceptibility is planned to be identified by the RI-ITG in a separate MRP document.[2] This chapter provides a brief summary of the important parameters for each age-related degradation mechanism for performing a preliminary screening and categorization, as described in Chapter 4.

2.1 PWR Internals Aging Mechanisms

2.1.1 Irradiation Embrittlement (IE)

Exposure to high-energy neutrons ($E > 1.0$ MeV) causes changes in the mechanical properties of the stainless steel and nickel-base alloys (including welds) used in PWR internals. Neutron irradiation increases yield and ultimate stresses, and decreases the ductility and fracture toughness of these PWR internals materials. A recent review of irradiation embrittlement (IE) provided the following conclusions:[18]

2.1.2 Stress Corrosion Cracking (SCC) [Excluding Irradiation Effects]

Stress corrosion cracking (SCC) occurs when the following conditions are present: 1) a tensile stress (both applied and/or residual stresses), 2) a corrosive environment, and 3) a susceptible material. SCC will not occur if any one of these three factors is eliminated. SCC can occur either as intergranular stress corrosion cracking (IGSCC) or as transgranular stress corrosion cracking (TGSCC), depending upon the environment/material combination. (Note: irradiation-assisted SCC is discussed separately in Section 2.1.3.)

PWR internals items that have exhibited SCC are highly-stressed Alloy A-286 fasteners and Alloy X-750 support pins. The Alloy X-750 SCC failures though are generally described as primary water stress corrosion cracking (PWSCC), which is a term traditionally used for IGSCC of nickel-base materials in PWR primary coolant systems.

In addition, Alloy 600 and similar nickel-base alloy (e.g., Alloy 182, Alloy 82) materials (should they exist in PWR internals) are also potentially susceptible to PWSCC. Although no SCC has been observed to date in PWR internals fabricated with austenitic stainless steel, a remote potential exists for SCC of PWR internals component items fabricated with austenitic stainless steels that may have been severely cold worked (e.g., >20%).

The observation of low temperature crack propagation (LTCP) with nickel-base alloy materials in PWRs has been described in several publications.[23-27] LTCP is characterized as a significant degradation in fracture toughness in low temperature hydrogenated water conditions at K-levels as low as 40 MPa√m (36 ksi√in) for nickel-base material. This effect is attributed to a hydrogen-induced intergranular cracking mechanism, a form of hydrogen embrittlement. The potential for LTCP occurring with stainless steel materials has not been demonstrated to date.

Austenitic stainless steels are generally susceptible to SCC in elevated temperature environments where halogen levels (e.g., chlorides and fluorides) are >150 ppb and/or dissolved oxygen is >5 ppb. During normal PWR operating conditions, when the materials are most susceptible to SCC, radiolytic dissolved oxygen concentration is maintained at <5 ppb by injecting externally-supplied hydrogen. The hydrogen addition serves to suppress radiolysis and to shift the electrochemical potential (ECP) into a regime in which cracking is not thermodynamically favored. During refueling outages, however, the primary water system will by necessity be exposed to oxygen when the reactor head is removed, if not before. Since this shift in environmental conditions from a highly reducing environment (hydrogen) to oxidizing conditions will cause a release of corrosion products (particularly Ni and Co-58), plants intentionally oxidize the primary water system during cooldown, while the material release can be removed by the letdown demineralizers. During plant cooldown, hydrogen peroxide is added to the primary coolant at temperatures typically <180°F (<80°C) to dissolve Co-58 and Ni out of the oxides so that they can be subsequently removed by the ion exchange resins prior to shutdown. The temperature limitation on hydrogen peroxide addition is two-fold, the primary reason being to limit exposure of the primary water system to an oxidizing condition while at temperatures high enough to cause aggressive SCC. The secondary reason for the temperature limit is that peroxide is more stable at lower temperature, allowing the plant to achieve a positive peroxide residual following the addition. At shutdown conditions, the primary coolant is exposed to air, which maintains the coolant under mildly oxidizing conditions. However, shutdown periods are relatively short and the temperatures are much lower compared with the time and temperatures of normal operation when oxygen levels are negligible. The above controls (i.e., hydrogen injection during operation and limiting oxygen exposure until Mode 5 during shutdown) have historically eliminated the potential for SCC of PWR internals components although continued surveillance should be practiced.

2.1.3 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Irradiation-assisted stress corrosion cracking (IASCC) is a degradation mechanism where materials exposed to neutron radiation become more susceptible to SCC with increasing fluence. A working threshold fluence of approximately 1×10^{21} n/cm^2 (E > 1.0 MeV) has been estimated from an extrapolation of PWR failure data.[28] However, this neutron fluence threshold could be conservative.

Operating experience and irradiated material test results suggest that IASCC may occur in the later stages of PWR internals life. One theory is that radiation-enhanced segregation of low temperature melting elements to grain boundaries causes the grain boundaries to be susceptible to accelerated attack. This radiation-induced segregation (RIS) is caused by diffusion of point defects (vacancies and interstitials) toward sinks such as dislocations, free surfaces, and grain boundaries, and by the difference in diffusion rates for different elements. Concurrently, the material is significantly hardened by irradiation. Although the exact mechanism of IASCC in PWRs is not yet known, both hardening and radiation-induced segregation (RIS) could play a role.[29] While the mechanical properties, such as yield strength and fracture toughness appear to saturate around 10 to 20 dpa of irradiation, IASCC susceptibility may continue to increase.

2.1.4 Thermal Aging (TA)

Thermal aging (TA), sometimes known as thermal embrittlement, is a time and temperature dependent process whereby a material undergoes microstructural changes leading to decreased ductility, and degradation of toughness and impact properties. This phenomenon is usually accompanied by an increase in yield strength, ultimate tensile strength, and hardness. Wrought austenitic stainless steels and nickel-base alloys are not subject to thermal aging at PWR operating temperatures.[30] For the conditions of PWR operation, the only materials in the PWR internals that are currently potentially susceptible to thermal aging are stainless steel welds, cast austenitic stainless steels (CASS), and precipitation-hardenable (PH) stainless steels.[31,32] CASS and PH materials are typically embrittled in the temperature range of 700 to 1000°F (371 to 538°C) within a short time.[33] For example, Charpy impact test data show that CASS thermal aging embrittlement reaches saturation after 2,600 hours at 752°F (400°C).[34] In addition for Type 17-4PH, Charpy impact test data show severe thermal aging embrittlement after 250 hours at 427°C (800°F).[35]

Different heats of CASS may exhibit different degrees of property degradation depending on the amount, size, and distribution of ferrite in the duplex austenitic/ferrite structure and the presence of carbides at the grain boundaries.[31] The following CASS alloys have been determined to be susceptible to loss of toughness:[37]

1. Centrifugal castings with greater than 20% ferrite
2. Static castings with molybdenum content less than 0.50% and ferrite greater than 20%
3. Static castings with molybdenum content greater than 0.50% and ferrite greater than 14%

In addition, austenitic stainless steel welds (e.g. Type 308 and 308L) have low levels of ferrite and molybdenum contents, such that it is expected that the effects of TA will be minimal because the welds are similar in structure to static castings.

The Nuclear Regulatory Commission (NRC) Staff has proposed the existence of a potential synergistic[1] effect of neutron irradiation on thermal aging. There are currently no data to prove or disprove this proposal. The WOG Materials Subcommittee (in collaboration with the MRP RI-ITG) is sponsoring tests to obtain PWR and test reactor irradiated material fracture toughness data from CASS component materials and austenitic welds with different amounts of thermal aging. These material samples will yield data in the near future that will shed light on the postulated synergistic effect.

2.1.5 Fatigue

Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized microstructural damage has been accumulated, crack initiation can occur at the most highly affected locations. Subsequent cyclic loading and/or thermal stress can cause crack growth. A brief description of the relevant fatigue-related degradation mechanisms is provided below.[38]

High-Cycle Fatigue

The most “classical” fatigue-related degradation mechanism is high-cycle (HC) fatigue. HC fatigue involves a high number of cycles at a relatively low stress amplitude (typically below the materials yield strength but above the fatigue endurance limit of the material). HC fatigue may be:

- Mechanical in nature, i.e., vibration or pressure pulsation or due to flow-induced vibration (FIV). FIV can induce HC fatigue in otherwise normally passive components merely through the interaction of flow adjacent to the component or within the system, establishing a cyclic stress response in the component. Thus, attention has to be directed to bolted

[1] The word synergistic, in this case, refers to the possibility that the effects of neutron irradiation and thermal aging could be greater than the sum of the effects from each mechanism considered individually.

connections subject to relaxation, particularly by irradiation creep. Additionally, power uprates are also of concern as an increase in flow may change the acoustical characteristics of the system and excite a HC mode where a resonant frequency is achieved.

- Thermally-induced due to mixing of cold and hot fluids where local instabilities of mixing lead to low-amplitude thermal stresses at the component surfaces exposed to the fluid.
- Due to combinations of thermal and HC mechanical loads.

Low-Cycle Fatigue

Low-cycle (LC) fatigue is due to relatively high stress range cycling where the number of cycles is less than about 10^4 to 10^5. To induce cracking at this number of cycles, the stress/strain range causes plastic strains that exceed the yield strength of the material, and the cycling causes local plasticity leading to more rapid material fatigue degradation. The stress cycling that contributes to LC fatigue is generally due to the combined effects of pressure and temperature changes that result during normal operation.

Thermal Fatigue

Thermal fatigue is caused by cyclic stresses resulting from changing temperature in a component or in the item attached to the component. Thermal fatigue may involve a relatively low number of cycles at a higher stress (e.g., plant operational cycles or injection of cold water) or due to a high number of cycles at low-stress amplitude (e.g., local leakage effects or cyclic stratification).

Environmental Fatigue

The term “environmental fatigue” refers to the reduction in fatigue “life” in a PWR environment compared to the “room temperature air” environment. Environmental fatigue involves two primary elements: i) the effects of a PWR environment on the overall fatigue life (as represented by either multiplying the fatigue usage factor by a secondary factor to account for environment or use of an environment-adjusted fatigue design curve), and ii) the potential accelerated growth of an identified defect due to the PWR environment. Environmental acceleration of fatigue crack growth is important in dispositioning detected/postulated flaws in a component to permit continued operation. At present, the fatigue curves in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code are based on data obtained from specimens tested in air and there is a significant effort underway to evaluate and incorporate the environmental effects.

The understanding of environmental fatigue is still evolving and is under considerable discussion in the technical community, Code bodies, and regulatory agencies. Laboratory data indicate that, for austenitic materials, the fatigue resistance is lower in deoxygenated primary water environments than in room-temperature air.[39] Code safety factors may not bound this difference in all cases. In addition, the effects of flow adjacent to affected items may reduce some of the environmental effects.

2.1.6 Stress Relaxation and Irradiation Creep (SR/IC)

The following paragraphs summarize the current knowledge on both thermal- and irradiation-induced creep and stress relaxation.[10]

Summary of Thermal Creep and Stress Relaxation

Creep is a thermally-activated process and is often used to refer to plastic strain produced under stress (stress or load does not have to be constant) at elevated temperature. However, thermal creep is usually more precisely defined as plastic strain under constant load or constant stress. At lower temperatures (below approximately half the melting point of a material), thermal creep is limited to the primary (transient) stage, which is considered insignificant as far as component geometry or distortion is concerned. Stress relaxation (SR), on the other hand is caused by the same mechanisms as creep, but has fixed total elongation instead of fixed load. Stress is reduced (i.e., relaxed) when the elastic strain is replaced by plastic strain.

The transient creep strain consists of anelastic strain and microplastic strain, which reach their limit in a short period of time. However, an estimated steady-state creep rate on the order of 5×10^{-13} sec^{-1} (tensile strain rate) should be considered for SR concerns over decades of operation at the temperature range pertinent to PWR internals. Although such creep strain is very small, it could cause additional SR not anticipated from short-term SR tests. Creep strain associated with volume change due to precipitation (thermal aging) is not considered to be a concern in the absence of irradiation for the materials and temperature range associated with PWR internals.

The available data show that SR appears to reach saturation in a short time (less than 1000 hours) with a maximum reduction of 10 to 20% of initial preload at PWR internals temperatures. However, this is based on tests that only lasted 1000 hours or less.[40] These tests have not taken into account the accumulated steady-state creep strain over a long operating life (nor the effects of cyclic loading).

Summary of Irradiation Creep and Stress Relaxation

Despite the amount of research work in the field of irradiation creep (IC) and SR over the past three decades, data pertinent to the PWR internals environment are limited, but the information being developed in the JOBB Program will provide considerable data in the near future. There could be an effect of void swelling (see Section 2.1.7) on IC and irradiation-enhanced SR under conditions pertinent to the PWR internals environment, but the data to date show a limited effect. Recent studies suggest that the low flux PWRs could experience void swelling. Such swelling in high fluence areas could give rise to differential swelling between materials, and therefore, a degree of reloading between bolted connections.[41,42]

For PWR internals, IC could be divided into two stages, the transient and the steady-state. Additionally, void swelling could also affect the creep rate, if it were significant. The transient creep is generally considered short and the contribution to the total creep strain small, but not necessarily insignificant. Recent IC experiments performed inside PWR fuel rods show less than

0.05% transient creep. Although a quantitative prediction often implicitly uses equations derived from the experimental data, the transient creep strain is often ignored in the steady-state creep strain.

Neutron flux causes SR to a higher degree than that caused by thermal SR at PWR temperatures. In general, there is a sharp initial reduction in preload stress corresponding to the transient creep. This is followed by a more gradual decline in stress, which does not appear to saturate, corresponding to steady-state IC. The lack of irradiation-enhanced SR data has led to empirical equations correlating SR with IC data. Such empirical equations are used to estimate SR before the onset of void swelling provided that material coefficients can be deduced from creep data on material with similar chemical compositions and thermo-mechanical conditions. This is a reasonable approach given that IC and irradiation-enhanced SR appear to be driven by the same process and differ only in that the former is stress/load driven and the latter displacement controlled. One quantitative post-void swelling SR study has been found that reports void swelling tends to accelerate SR in the same manner as it affects IC.[43]

2.1.7 Void Swelling (VS)

Irradiation-induced swelling, also known as void swelling (VS), occurs in austenitic stainless steels and nickel-base alloys during neutron bombardment. Voids grow due to the migration and condensation of vacancies on the void nuclei (which can be gas bubbles) leading to an increase in material volume.[10] This phenomenon is dependent on temperature, neutron flux, and neutron fluence. Swelling is typically a concern for fast reactors at temperatures above 350^{o}C (662^{o}F). Under PWR conditions a high density of helium bubbles are formed without any macroscopic increase in volume. Among these bubbles, voids have been observed. The maximum swelling observed to date has been 0.24%.[10] The known interactions between materials, chemical composition, stress, fluence, and temperature from fast reactors are summarized in MRP-50,[10] suggesting that VS in PWR internals could become significant in localized areas subject to temperatures approximately > 320°C (608°F) and doses approximately >80 dpa (approximately >5 X 10^{22} n/cm^2, E > 1.0 MeV). To date, no known PWR components have exhibited significant swelling.

A large uncertainty currently exists in defining the transient regime duration before significant VS begins to occur for PWR internals materials. To date, with up to 30 years, under relevant conditions, no practical problem has been observed. However, there is one additional aspect of swelling to consider. If swelling exceeds 10%, the tearing modulus is expected to fall to low levels (<10) based on fast reactor data and the material becomes exceptionally brittle. The effect of this embrittlement would be most severe after the reactor has cooled down and the material is at temperatures below 200°C (392°F).[10]

2.1.8 Wear

Wear is the loss of material, generally measured as the rate of removal of surface material, caused by the relative motion between adjacent metal surfaces or by the action of hard, abrasive particles in contact with a metal surface. Mechanical wear is observed in bolted or clamped joints where relative motion is not intended, but which occurs due to the reduction or loss of preload. Flow-induced vibration (FIV) is also known to be a cause of wear through intermittent contact of adjacent metal surfaces. A limited number of PWR internals items, such as thimble tubes, are subject to such relative motion.

2.2 Effects of Aging Degradation in PWR Internals

The effects of aging degradation in PWR internals can be characterized in terms of cracking (which includes crack initiation and growth), reduction of fracture toughness, loss of mechanical closure integrity, loss of material, or changes in dimension. The various age-related degradation mechanisms that could result in these effects are summarized in Section 2.1. The susceptibility of the PWR internals items to these age-related degradation mechanisms is dependent upon such factors as material composition, manufacturing process, product form, the operational environment (i.e., neutron fluence, neutron flux, temperature, and water chemistry), and maintenance history. A summary of the relationships between age-related degradation mechanisms and effects is provided below.

2.2.1 Cracking

Age-related degradation mechanisms that may lead to cracking of the PWR internals items include IASCC, PWSCC/LTCP, and fatigue (HC and Environmental). Cracking due to SCC is not expected to be a significant aging mechanism for the PWR internals because of the reactor coolant chemistry controls in place, as required by plant Technical Specifications. However, some cracking mechanisms could potentially occur because of other key factors. For example, IASCC is considered a potential age-related degradation mechanism in PWRs for stainless steel and nickel-base alloys that will experience high fluence exposures due to their close proximity to the core. The currently identified potential materials and age-related degradation mechanisms of concern for cracking are as follows:[38]

- IASCC – Wrought stainless steel, stainless steel welds, CASS, and wrought nickel-base alloys
- PWSCC/LTCP – Wrought stainless steel, stainless steel welds, CASS, and wrought nickel-base alloys
- Fatigue (HC and Environmental) – Wrought stainless steel and CASS alloys

2.2.2 Reduction of Fracture Toughness

Age-related degradation mechanisms that may lead to a reduction of fracture toughness of the PWR internals items include TA, IE, and VS. A consequence of reduced fracture toughness is a reduction in a materials critical crack size. Aging management of items with reduced fracture toughness need to rely upon observations of flaw length and the magnitude of stress/loading. Of particular importance is the issue of VS. If swelling exceeds 10% (based on existing fast reactor wrought stainless steel data), the tearing modulus is expected to fall to near-zero and the material becomes exceptionally brittle, being most severe after the reactor has cooled down and the material is at temperatures below about 200°C (392°F). The currently identified potential materials and age-related degradation mechanisms of concern for reduction of fracture toughness are as follows:[38]

- TA – Stainless steel welds and CASS alloys
- IE – Wrought stainless steel, stainless steel welds, CASS, and wrought nickel-base alloys
- VS – Wrought stainless steel, stainless steel welds, and wrought nickel-base alloys

2.2.3 Loss of Mechanical Closure Integrity

Loss of mechanical closure integrity is caused from SR/IC of those component items (i.e., bolting) where maintaining a preload is important to the structural integrity function of the PWR internals. Neutron fluence and the degree of preloading are the key parameters. Thus the identified potential materials and age-related degradation mechanisms of concern for loss of mechanical closure integrity are:[38]

- SR/IC – Wrought stainless steel and wrought nickel-base alloys

2.2.4 Loss of Material

The only known aging degradation mechanism that results in loss of material is from wear due to mechanical abrasion in circumstances where items are physically in contact and able to move, either by design or due to FIV. Plant operating conditions usually determine the severity of wear. Loss of material due to wear is not considered a potential aging effect for bolted items provided the bolts continue to maintain sufficient preload, as discussed in Section 2.2.3, or do not sever as a result of cracking. Therefore, the identified potential materials and age-related degradation mechanism of concern for loss of material is:[38]

- Wear – Wrought stainless steel, CASS, and wrought nickel-base alloys

2.2.5 Changes in Dimension

Changes in dimension due to VS could lead to loss of component function if the required clearances of the PWR internals items cannot be maintained for: 1) the orientation, guidance, and protection of the control element assemblies, 2) distribution of the reactor coolant flow to the reactor core, or 3) support, guidance, and protection for the in-vessel core instrumentation. If VS does occur in the PWR internals, it is most likely to be a localized phenomenon in regions of peak temperature and neutron fluence. If sufficient amounts of VS were to occur during the period of extended operation, the dimensional changes would need to be managed. The currently identified potential materials and age-related degradation mechanism of concern for change in dimension is as follows:[38]

- VS – Wrought stainless steel, stainless steel welds, CASS, and wrought nickel-base alloys

3
PWR INTERNALS DESIGNS AND SAMPLE COMPONENT TEMPLATE

The purpose of this chapter is to provide an understanding of the basic PWR internals designs and materials so that management of the known age-related degradation mechanisms will provide assurance that the design requirements are satisfied. This information focuses on the generic aspects of PWR internals design. However, it must be recognized that there are features which are not common to all vendor designs and, in these cases, specific design features will be identified. Figure 3-1 shows the PWR internals structural assembly groupings: the upper internals assembly, the core support assembly, and the lower internals assembly. Figures 3-2, 3-3, and 3-4 illustrate the typical features of the Westinghouse, Combustion Engineering, and Babcock & Wilcox designed PWR internals. The PWR internals are designed to serve several functions:

1. Provide support and orientation of the reactor core (i.e., fuel assemblies).
2. Provide support, orientation, guidance and protection of the rod cluster control assemblies (RCCA) in Westinghouse plants. These are referred to in the Combustion Engineering and Babcock & Wilcox plants, respectively, as control element assemblies (CEA), and control rod assemblies (CRA).
3. Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
4. Provide a passageway for support, guidance, and protection for in-vessel/core instrumentation.
5. Provide a secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.
6. Provide gamma and neutron shielding for the reactor vessel.
7. Provide positioning and support for the reactor vessel surveillance capsules (most CE and B&W plants excepted) and internals vent valves (B&W plants only).

Similarity exists in all PWRs in the overall means of structural support and guidance, and in the flow of coolant (excluding direction) through the reactor vessel and internals. While there are some differences in the particular components used to achieve core support, PWR internals can generally be divided into two main subassemblies: 1) a lower assembly which consists of a lower support structure attached to a core barrel, and 2) an upper assembly which is installed after the lower assembly and fuel assemblies have been installed.

The fuel assemblies rest on the lower support structure of the lower assembly, which transmits the resulting load to the core barrel and then to the core barrel flange, which rests on the reactor vessel flange. The upper assembly is clamped under the reactor vessel head flange and provides the upper structure interface with the fuel assemblies. During refueling operations, the upper assembly is removed from the reactor vessel to allow access to the fuel assemblies. This provides an opportunity to perform inspections of the upper internals components. The core barrel also provides a flow boundary for the reactor coolant as illustrated in Figure 3-1. When the primary coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the vessel. The flow then enters the lower plenum between the bottom of the lower assembly and the vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper assembly region and then proceeds radially outward through the reactor vessel outlet nozzles. The perforations in the various components, such as the lower support structure, control and distribute the flow to the core. In some PWR internals designs, a small amount of bypass flow is allowed to enter the vessel head plenum for cooling purposes.

Before the development of the ASME B&PV Code requirements specifically applicable to PWR internals, the design of PWR internals was based on criteria specific to each vendor. However, Section III of the ASME B&PV Code was used as a guideline for the PWR internals design criteria. PWR internals whose contract dates followed the issuance of the 1974 Edition the ASME B&PV Code Section III, were designed to satisfy Subsection NG, Core Support Structures. Among the requirements contained in Subsection NG are rules for fatigue evaluation and categorization of PWR internals loads. The rules for elevated temperature service of metals whose temperatures exceed the ASME B&PV Section III allowables are in Code Case N-201.

Materials for the PWR internals were chosen to meet these Code requirements. A sample of generic PWR internals materials of construction is given in Appendix A. Each vendor chose slightly different materials to meet the particular design needs.

Table 3-1 provides a sample template format to be used in tabulating the PWR internals components. This table should be completed for each vendor type of design to identify the specific materials used for each component along with the operating parameters, such as temperature and fluence. For categorization purposes, conservative upper bound values are less useful than narrow range estimates. If available, detailed temperature and fluence ranges should be provided for each item, as a function of location on the component or distance from the core. This information is crucial input for the next steps to categorize and screen each component for significance.

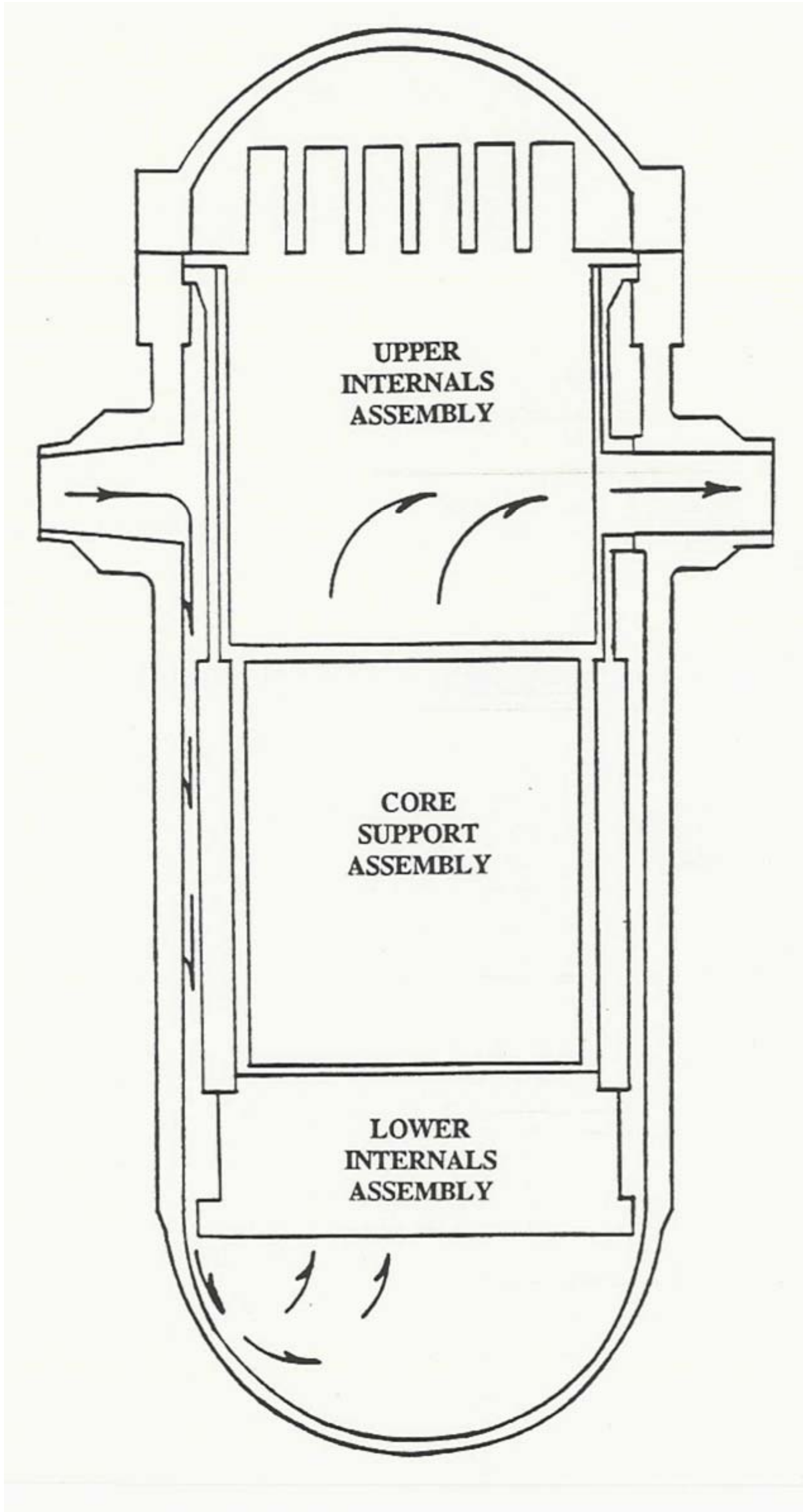

Figure 3-1
PWR Internals Structural Assembly Groupings [45]

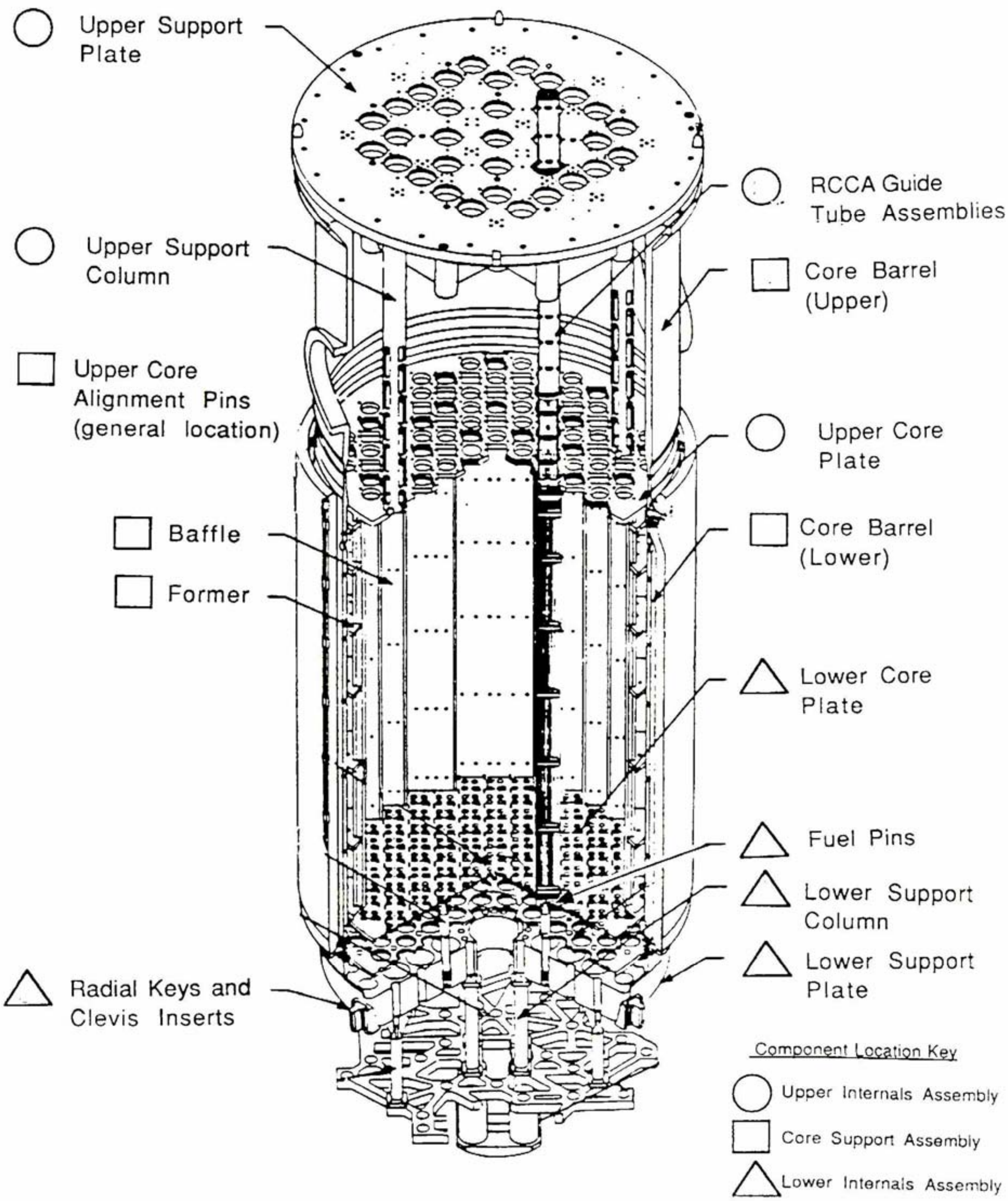

Figure 3-2
Westinghouse Designed Internals System [45]*

*** Sample configuration - unique variations may exist for individual plant designs**

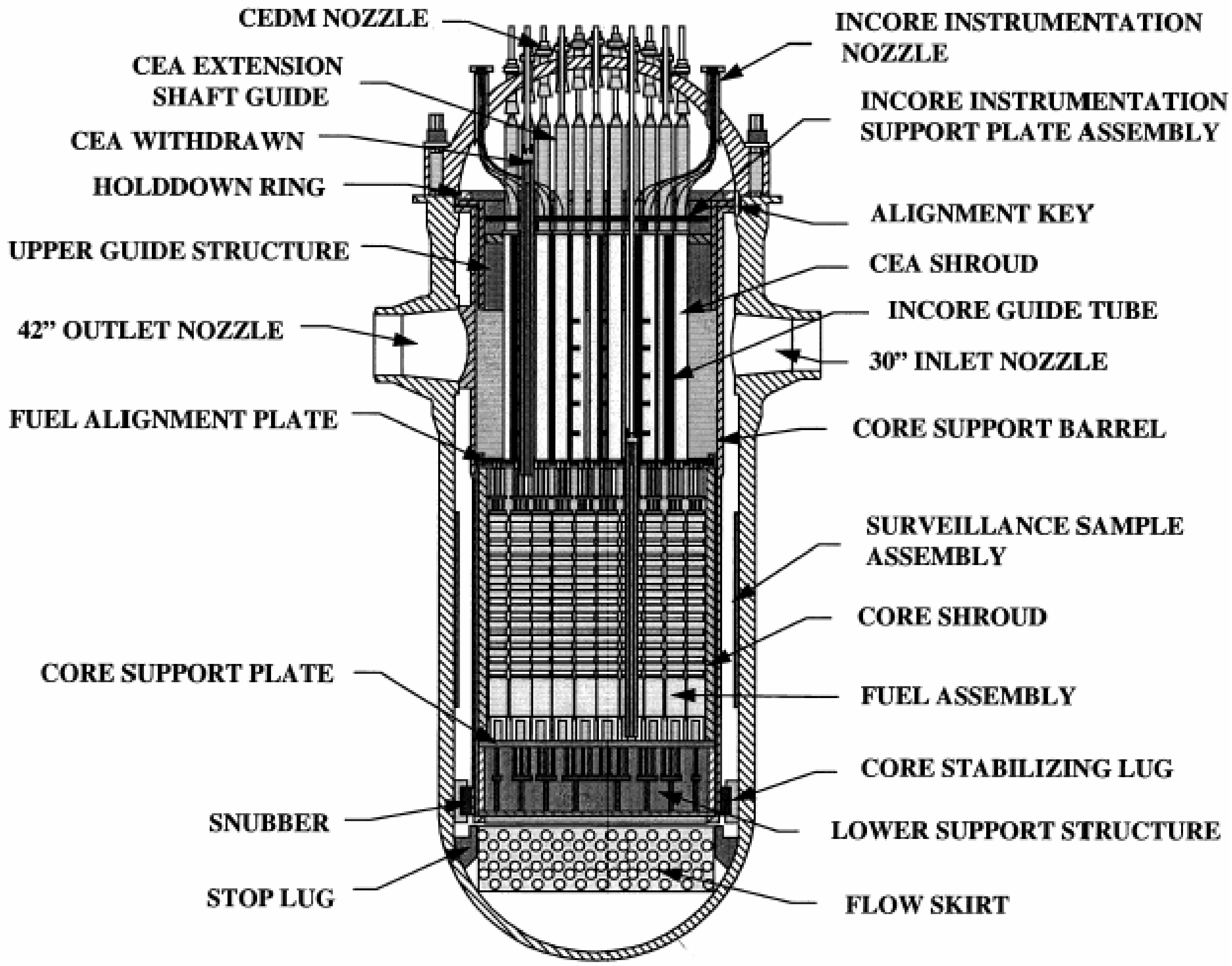

Figure 3-3
Combustion Engineering Designed Internals System [45]*

*** Sample configuration - unique variations may exist for individual plant designs**

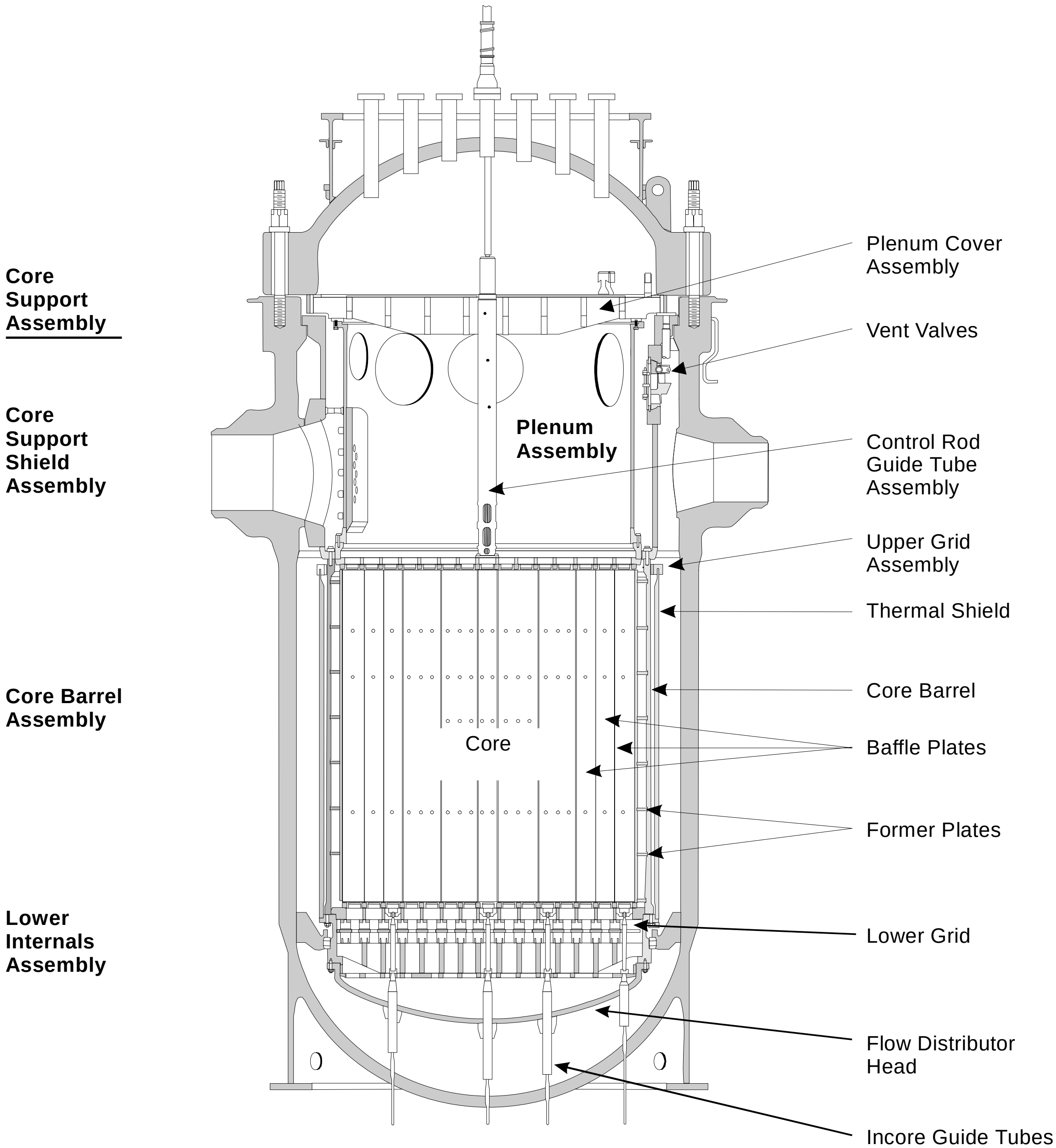

Figure 3-4
Babcock & Wilcox Designed Internals System [45]*

*** Sample configuration - unique variations may exist for individual plant designs**

Table 3-1
Sample Template of PWR Internals Materials of Construction and Operating Parameters

Item	Material	Specification Number	Type or Grade/Class	Product Form	Typical Temperature (°F)	Typical Fluence (n/cm^2, E > 1.0 MeV)
Upper Internals Assembly						
Upper Support Plate	Stainless Steel	SA-240	Type 304	Plate	< 600°F	$< 10^{21}$
Upper Support Column	Stainless Steel	SA-439	Type 304	Plate	< 600°F	$< 10^{21}$
Upper Support Column Bolts	Nickel Alloy	SA-637	Alloy X-750	Bar	< 600°F	$< 10^{21}$
(Sample Input …………………… Actual materials information and parameters to be completed for each vendor design type)						

4
PWR INTERNALS COMPONENT CATEGORIZATION PROCESS

A key aspect of the aging management strategy using inspections is identifying the PWR internals components of greatest significance due to the effects of age-related degradation. In the screening and categorization for significance, aging susceptibility factors are examined to identify those conditions that could contribute to the aging effects identified in Section 2.2. A prioritization of the significance of aging degradation in PWR internals is proposed for the screening process such that the significance is a combination of the susceptibility to aging degradation and the potential for loss of function with materials degradation.

$$\text{Significance} \cong \begin{bmatrix} \text{Potential for Aging Based} \\ \text{on Susceptibility Factors} \end{bmatrix} \text{x} \begin{bmatrix} \text{Potential for Loss of Function} \\ \text{With Materials Degradation} \end{bmatrix}$$

Susceptibility Factors	Functionality Assessment Results
* Fluence	
* Stress	
* Temperature	
* Material	

The aging susceptibility factors must be related to quantifiable criteria such as fluence, temperature, stress, material type and composition, etc. to be meaningful for the determination of significance. Because significance is a relative measure of aging degradation, the actual criteria are somewhat subjective but are intended to be indicators of the level of degradation that may be tolerated without harm to the component function.

Four categories of components are defined below for classification of the significance to aging degradation effects.

Category A

Category A components are those for which aging effects are below the screening criteria, so that aging degradation significance is minimal. Typically, only the required ASME B&PV Code Section XI Examination Category B-N-3 ISI visual examinations (VT-3) will be performed on these components to assess potential aging effects.

Category C

Category C PWR internals components are those "lead" components for which aging effects are above screening levels, which have moderate or high susceptibility to degradation, and have not yet been demonstrated to be sufficiently tolerant to remain functional relative to aging degradation significance. Enhanced inspections (e.g., Enhanced VT-1, UT, etc.) and/or surveillance sampling will typically be warranted to assess aging effects and verify functionality of these components.

Category B

Category B includes those PWR internals components that are moderately susceptible to the aging effects, such that the effects on function cannot easily be dispositioned by screening and are not "lead" components. Category B components may require additional evaluations to be shown tolerant of the aging effects with no loss of functionality (i.e., damage tolerant). These components are candidates for a better than VT-3 quality remote visual examination. An example of one such technique, termed VT-3/VT-1, could be implemented; implying that the remote examination may satisfy the standoff distance and character recognition requirements for a VT-1 visual examination.

Category B'

Category B' components are those "lead" components that can be shown to be tolerant of the aging effects through a functionality assessment. Aging degradation significance for these PWR internals components are considered to be manageable using a better than VT-3 quality remote visual examination (e.g., VT-3/VT-1), but may also be candidates for an expanded inspection program.

Given these rough categories for grouping or "binning" of the PWR internals components, a process was developed to identify the aging degradation significance as a key step in developing inspection guidelines for PWR internals. The steps in this process are shown in Figure 4-1, and are performed as follows:

1. For each component and potential aging mechanism, perform an initial screening based on fluence, stress, temperature, material, etc. to eliminate those components that can easily be demonstrated to be below the screening criteria. These become Category A components that can be managed using standard ASME B&PV Code Section XI Examination Category B-N-3 visual inspections. These examinations are intended to provide general condition assessment.

2. For the remaining components, use existing information to identify "lead" components for each potential aging degradation mechanism. Any component that is deemed to be a lead component for ***any*** aging degradation mechanism is placed into Category C. These components will remain in Category C unless further evaluation can be performed to reduce the significance for aging management. For example, a functionality or flaw tolerance

evaluation may show that the component does not require the rigor of the aging management inspection programs applicable to Category C components.

3. All other components are placed into Category B. These components do not satisfy the screening criteria for the aging degradation mechanism effects, and are not considered to be "lead" components for any aging degradation mechanism effect.

4. After the initial classification process is complete, key components from Category C will be selected for further evaluation. Combinations of functionality assessment, flaw tolerance evaluation, and safety evaluation may be used to show that these components are damage tolerant and can be moved to Category B'. Those that are moved to Category B' represent candidates for expanded inspections pending the outcome of the Category C component inspection results.

5. This process is continued until the most cost-effective aging management/inspection plan is achieved.

The Figure 4-1 categorization process described above effectively translates the degradation significance concept formula on page 4-1 to a binned measure of inspection importance, as illustrated in Table 4-1. The table has ordinates of susceptibility and functionality, and specifies graduated levels of inspection importance depending on the combination of these factors. Inspection importance can then be used as a basis for the specification of the actual combination of inspection examination techniques, coverage, sampling, and frequencies.

The table is based on *a-priori* estimates of susceptibility and is not by itself a complete inspection plan, since it is intended chiefly as a guide to optimizing the screening and functionality evaluations required for developing the actual inspection plan. The table is premised on a limited rate of emergence of degrading effects. If inspections are performed in accordance with this table and only isolated flaws are found in lead items, then the premise is confirmed and inspections are complete. If significant flaws are found, then additional inspections may be required to map and characterize the full extent and its impact on functionality.

Details about the use of Figure 4-1 in the categorization of PWR internals are provided in Chapter 5. Additional details on inspection techniques and strategy for managing aging effects in PWR internals are provided in Chapter 6 and in a separate report on PWR Internals Inspection and Flaw Evaluation Approaches.[3]

Figure 4-1
Process for Categorization of PWR Internals Components

5 APPROACHES FOR AGING DEGRADATION CATEGORIZATION AND FUNCTIONALITY OF COMPONENTS

5.1 Screening and Categorization for Significance

Chapter 4 described the categorization process for prioritizing the significance of aging degradation in PWR internals components. The usefulness of this categorization approach is that the lead components can be identified, evaluated in more detail for functionality, and included in an inspection program to assure continued function of the components for the long term.

Screening criteria based on combinations of fluence exposure, operating temperature, material, stress level, etc. are needed to establish the susceptibility factors. A summary of the important parameters for each age-related degradation mechanism is provided in Chapter 2 for performing a preliminary screening of PWR internals components. More refined threshold and screening criteria and the technical bases for the criteria will be developed in a separate MRP report.[2]

There is a distinct difference between threshold values for measuring the onset of aging effects and screening values for evaluating the significance of age-related degradation mechanisms. The following are working definitions:

Threshold Value - The level of susceptibility when an aging effect is first quantifiable

Screening Value - The level of susceptibility when an aging effect may be significant to functionality.

Quantification of the screening criteria will require knowledge of the specific aging mechanisms, some engineering judgment, and possibly empirical relations where data may be lacking. The screening criteria will be used by those organizations performing the screening and categorization of components.

5.2 Examples of Screening for Age-Related Degradation

A sample screening of components for one vendor design type may be helpful in demonstrating the process for identifying "lead" components, as shown in the following examples.

Example A – Screening of Components for Irradiation Embrittlement

A sample of lead components exposed to high neutron irradiation fluence levels are as follows:

Example B –Screening of Components for IASCC

Factors contributing to IASCC are high neutron fluences, tensile loading for some bolts approaching the material yield strength and temperature increases due to gamma heating. As a result, the baffle/former bolts are expected to be leading locations for IASCC of the internals.

The screening of PWR internals components for susceptibility to cracking due to IASCC should be performed on a plant-specific basis, unless it can be shown that the screening results would be bounded by fleet screening evaluations. There also may be sufficient similarities within individual vendor designs, such that a generic screening of components on a design-specific basis can be helpful as a first step in identifying lead components. This particular example for reactor internals components includes wrought and forged austenitic stainless steels. For this initial screening assessment, the most susceptible stainless steel reactor internals are separated into 2 groups of relative IASCC susceptibility.

Example C – Screening of Components for Thermal and Irradiation Embrittlement

With most CASS components under compression or very low tensile loads, lead components are identified by a combination of moderate fluence and either occasional or sustained tensile stress.

Example D – Screening of Components for Stress Relaxation

In general, baffle/former bolts are the most likely components to exhibit significant relaxation due to the combination of high neutron fluence and high temperatures experienced by these bolts.

Example E – Screening of Components for Void Swelling

In general, baffle/former bolts and plates are the most likely components to exhibit early indications of void swelling due to the combination of high neutron fluence and high temperatures experienced by these bolts.

5.3 Example of Component Screening for Functionality

Screening of components based on functionality analysis is also an option for managing aging effects in PWR internals, as shown in the Figure 1-1 framework.

For the purpose of screening, functionality is defined as follows: a component is considered to be functional if degradation to it or any of its subcomponents a) is insufficient to prevent the component from performing its required functions, and b) is insufficient to prevent any other component from performing its required functions.

Some preliminary assessments of component functionality have been already been considered by the vendors and utilities during license renewal evaluations. These simple functionality assessments may disposition some of the identified components requiring further examination and aid in the selection of components needing more detailed evaluations of functionality.

These examples are provided for illustration only.

Examples of Component Screening for Functionality for Westinghouse Designed Internals

5.4 Safety Evaluation of PWR Internals

If necessary for regulatory approval, age-related degradation of PWR internals may need to be examined for safety significance. The following three basic criteria must be satisfied to ensure that a nuclear safety concern does not exist (10 CFR Part 21, Section 21.3):

1. The integrity of the reactor coolant pressure boundary is maintained

2. The reactor must be capable of being shutdown and maintained in a safe shutdown condition, and

3. The capability must exist to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to 10 CFR Part 100 guidelines.

The above criteria must be demonstrated to be met under the current licensing basis. The effects of cracking or reduction of toughness in PWR internals could have safety significance if function cannot be maintained. Specifically, a failure to maintain core cooling during a design basis accident would violate the above criteria 2.

In addition to screening for functionality, components can also be screened on the basis of safety significance. This is the initial approach taken by the BWRVIP, and the evaluation procedure is referred to as a ***component safety assessment***. The procedure involves:

- The physical description of each major internals component;
- The functional description and, in particular, the safety functional description of each major internals component;
- A ***qualitative*** assessment of the consequences of significant age-related degradation effects on the performance of these functions, including – in some cases – narrative forms of consequence event trees leading to loss of function; and
- Any operating history and/or regulatory actions that confirm the assessment conclusions.

The first two steps in this safety assessment procedure, plus the fourth step, can be found in the Owners Group generic assessments (e.g., see References 16 and 21) and in the PWR License Renewal Industry Report.[45] The third step provides a method for screening out components for which the consequences of severe age-related degradation do not compromise safety function. A similar approach, based on ***quantitative*** assessment of the effects of age-related degradation on component safety function, has been proposed for the assessment of PWR internals components. That approach is referred to in this report as a ***component or system functionality assessment***. A safety assessment would further demonstrate that the function of the component(s) could be met even with the age-related degradation.

6
APPROACH FOR AGING MANAGEMENT OF PWR INTERNALS USING INSPECTIONS

This chapter of the report addresses the third step of the framework and strategy given in Chapter 1, which is identifying appropriate inspections to assure structural integrity of PWR internals components. In addition, this third step, although possibly comprising a wide variety of aging management program elements, will define the term “inspections” to include surveillance, testing, and monitoring elements, as well as in-service examinations.

The discussion that follows will take advantage of information available in existing regulatory documents that describe the requirements for such “inspection” program elements. For example, the Generic Aging Lessons Learned (GALL) Report [48] and the NRC Standard Review Plan [49] describe how “inspections” may be used for detection of aging effects in PWR internals. In particular, for detection of aging effects it is stated that:

> "Detection of aging effects should occur before there is a loss of the structure and component intended function(s). The parameters to be monitored or inspected should be appropriate to ensure that the structure and component intended function(s) will be adequately maintained for license renewal under all CLB design conditions. This includes aspects such as method or technique (e.g., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure the timely detection of aging effects."

The example given in Figure 6-1 shows how appropriate visual and volumetric examinations can be used to manage all possible aging effects for the range of potentially susceptible PWR internals components. The process relies on the categorization step described earlier to identify the most-affected or “lead” components. Regardless of the aging management program elements selected, the PWR internals inspections should be tailored for plant-specific aspects using the best available knowledge of the aging phenomena. The inspections should be structured to meet the key Aging Management Program attributes defined in the GALL Report.[48] These are described in Appendix B.

The Standard Review Plan [49] further states that the inspection method or technique and frequency may be linked to plant-specific or industry-wide operating experience. When sampling is used to inspect a group of components, the basis should be provided for the inspection population and inspection size. The samples should be biased toward locations most susceptible to the specific aging effect of concern in the period of extended operation. The inspection population should be based on such aspects of the components as a similarity of materials of construction, fabrication, procurement, design, installation, operating environment, or aging effects.

Appropriate "inspection" methods would be employed for managing the aging effects of concern (e.g., cracking). A brief discussion of the use of one-time or periodic surveillance, testing, monitoring, and examinations for managing aging effects in PWR internals is provided here. More details are provided in a separate report on PWR Internals Inspection and Flaw Evaluation Approaches.[3]

With regard to periodic examinations, ASME B&PV Code Section XI Subsection IWB Examination Category B-N-3 provides in-service examination requirements for accessible surfaces of removable PWR internals classified as core support structures (Class CS). However, the NRC staff has determined that Examination Category B-N-3 is inadequate, in part, as a program -- or an element of a program -- for managing ***some*** of the effects of aging during the license renewal term. In particular, deficiencies in the capability of Examination Category B-N-3 to manage the effects of cracking have been cited in the GALL report.[48] The implication is that detection and length sizing of crack-like indications using VT-3 visual examination is subject to uncertainty and potentially significant error.

In view of these perceived deficiencies, the EPRI MRP RI-ITG is considering that, for PWR internals component locations considered to have potentially significant age-related degradation effects, enhanced examinations (e.g., enhanced visual and ultrasonic) be used for aging management of the lead (i.e., Category C) components. For component locations initially screened out (Category A) the existing ASME B&PV Code Examination Category B-N-3 visual examination requirements would continue to be adequate. The use of augmented examinations will also be used for management of non-lead component locations (Category B and Category B'). For example, management could include one-time or periodic examination of a suitable percentage of the population. These categories also represent candidate components (or locations) for an expanded inspection program.

Of these four categories, requirements for Categories C, B, and B' represent new program elements. The use of enhanced inspection methods and location prioritization is similar to the approach used by the BWRVIP for managing degradation effects in BWR vessel internals.[50,51] Guidelines for future inspections of PWR internals are to be developed under the EPRI MRP.

Because of the uncertainty with respect to in-situ volumetric examination and the inadequacy of Enhanced VT-1 or VT-3 visual examination techniques for baffle/former bolts, prudence dictates that another program be defined to manage aging degradation. One potential approach would be to establish a surveillance program for the baffle/former bolts. The objective would be to provide measurements on selected baffle/former bolts removed periodically from operating plants to gain a better understanding of various degradation mechanisms (e.g., IASCC susceptibility, VS, and SR/IC). Similar testing and evaluation has already been performed on bolts removed from three U.S. operating plants. Another potential approach would be to manage the age-related degradation through evaluation of the research data (e.g., IASCC, VS, and SR) being generated through industry-sponsored programs.

Options should also be considered for early detection and, if possible, mitigation of possible aging degradation. Early detection of degraded components relies on related plant experience and modeling predictions of the aging mechanisms. An example of early detection of cracking is performing inspections (during the current operating license) and loose parts monitoring. An example of mitigation of cracking is provided through implementation of reactor coolant chemistry control. As new data and information are gathered on the age-related degradation mechanisms, it should be factored into the overall assessment of the leading locations for plant evaluation.

Repair/replacement options may also be considered as part of an overall program for aging management of PWR internals. For example, determination and replacement of the minimum pattern of baffle/former bolting required to ensure structural integrity of the baffle assemblies may eliminate the need for repeated inspections of these susceptible components. In addition, improved bolt design and/or materials selection may also reduce the significance of those components that have been repaired or replaced.

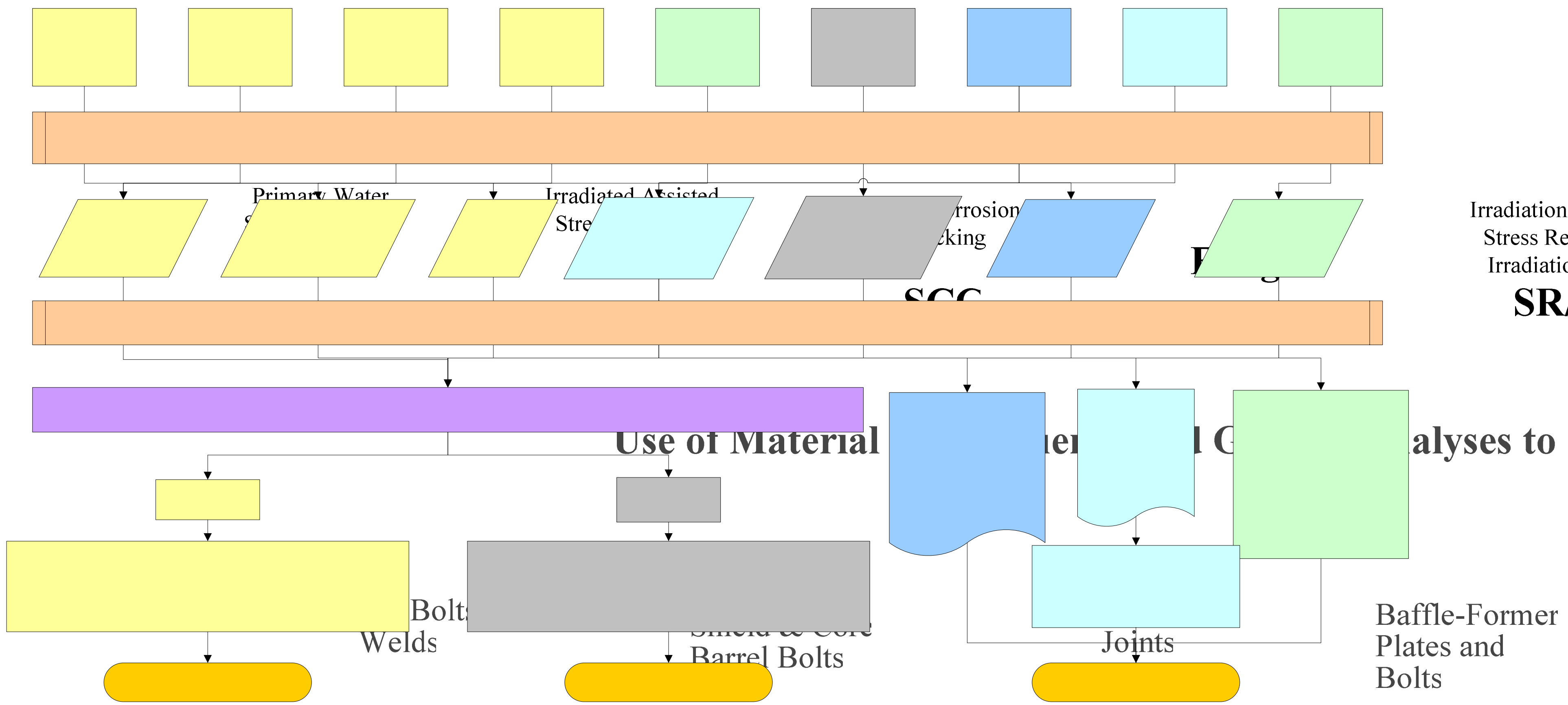

Figure 6-1
Example of PWR Internals Aging Management Strategy Using Inspections

7 SUMMARY

This report describes a framework and strategy for developing guidelines for managing the effects of aging in PWR internals. The overall framework is illustrated in Figure 1-1, which consists of three main strategies: a) screening and categorization, b) functionality and safety evaluations, and c) surveillance, testing, monitoring, and examination.

The first strategy uses knowledge of PWR internals design, materials and operating conditions to categorize the PWR internals components into groupings that depend on their susceptibility to age-related degradation. The most-affected components are defined in this report to be "lead" components, and are placed into Category C. As shown in the bottom portion of Figure 6-1, the third strategy in this report calls for augmented inspections, surveillance, and possible repair/replacement for these lead components. The second strategy is based on the recognition that functional assessments and other generic evaluations may reduce the lead component populations, thereby re-categorizing a Category C PWR internals component into Category B', as shown in Figure 4-1. The second strategy has another element that permits a Category B PWR internals component (moderate susceptibility) to be re-categorized to Category A (low susceptibility), based on functionality and safety evaluations. The Category B' population is available as a group of components that can be used to augment the population of lead components, should such augmentation be required. Such a need for an augmented population of lead components will depend on the findings from the enhanced aging management programs of inspection, surveillance, and repair/replacement. An important feature of this overall framework is that less-affected components (Category A) will be subject only to existing aging effects management program elements, whereas the Category B grouping is subject to augmented examinations, and the Category B' grouping is potentially subject to the additional requirements for the lead component population.

The preliminary inspection and surveillance guidelines introduce the following new program elements:

1. Enhanced examination of "lead" (Category C) components;

2. Enhanced examination of a technically appropriate scope of "non-lead" components (Categories B and B'); and

3. An integrated surveillance program for baffle/former bolting, in order to accumulate information that can be used to manage in-situ inspections of components.

The overall framework and the three strategies also depend on results from ongoing and future research programs, whether sponsored by the EPRI MRP RI-ITG, various owners groups, the

BWRVIP, or elsewhere. Additional data under PWR conditions are expected in a number of areas, including tensile behavior, crack initiation and crack growth characteristics, fracture toughness data, and void swelling measurements. These data may cause the initial categorizations to be modified.

8 REFERENCES

1. Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-135) - EPRI Report 1011127 (2005).

2. Materials Reliability Program: PWR Internals Aging Degradation Mechanism Screening and Threshold Values, EPRI report, Palo Alto, CA (under development).

3. Materials Reliability Program: PWR Internals Inspection and Flaw Evaluation Approaches, EPRI report, Palo Alto, CA (under development).

4. Analysis of Baffle Former Bolt Cracking in EDF CPO Plants (MRP-03), EPRI TR-112209, 6/1999.

5. Inspection and Replacement of Baffle to Former Bolts at Point Beach-2 and Ginna, EPRI TR-114779, 2/2000.

6. EPRI Baffle Bolt Project Summary, AP-114779-CD, 4/2000.Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51), EPRI Report 1003069, 11/2001.Determination of Operating Parameters of Extracted Bolts (MRP-52), EPRI Report 1006075, 10/2001.

9. *In-Situ* NDT Measurements of Irradiation-Induced Swelling in PWR Core Internal Components -- Phase 1: Testing of Unirradiated Surrogate Material, EPRI Report 100658, 10/2001.

10. Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50), EPRI Report 1000970, 11/2001. Strategies for Management of Aging Effects in PWR Reactor Vessel Internals (MRP-62), EPRI Report 1006582, 2/2002.

12. Characterization of Type 316 Cold-Worked Stainless Steel Highly Irradiated under PWR Operating Conditions (MRP-73), EPRI Report 1003525, 8/2002.

13. A Review of Thermal Aging Embrittlement in Pressurized Water Reactors (MRP-80), EPRI Report 1003523, 5/2003.

14. Stress Corrosion Cracking of High Strength Reactor Vessel Internals Bolting in PWRs (MRP-88), EPRI Report 1003525, 8/2002.

15. A Review of the Cooperative Irradiation Assisted Stress Corrosion Cracking Research Program (MRP-98), EPRI Report 1002807, 12/2003.

16. Strategies for Managing Aging Effects in PWR Vessel Internals – Interim Update (MRP-99), EPRI Report 1007848, 12/2003.Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples - Material Certification, Fluence, and Temperature (MRP-128), EPRI Report 1008202 and - Tensile and SSRT Testing (MRP-129), EPRI Report 1008205, 11/2004.

18. Materials Reliability Program: A Review of Radiation Embrittlement for Stainless Steels for PWRs (MRP-79) - Revision 1, EPRI Report 1008204, 9/2004.

19. Joint Owners Baffle Bolt Program (JOBB-CD), Version 4.12, EPRI Report 1008206, 11/2004.

20. *In-Situ* NDT Measurements of Irradiation-Induced Swelling in PWR Core Internal Components – Phase 2: Testing of Irradiated Materials, EPRI Report 1009762, 4/2004.

21. *In-Situ* NDT Measurements of Irradiation-Induced Swelling PWR Core Internal Components – Phase 3: Correlation of Void Swelling and Material Properties of Austenitic Steels, EPRI Report 1009763, 4/2004.

22. "Pressurized Water Reactor Vessel Internals License Renewal Industry Report," Revision 1, EPRI TR-103838, July 1994, EPRI, Palo Alto, California.

23. Mullen, J.V., and Parrington, R.J., "Stress Corrosion of Alloy 600 Weld Metal in Primary Water," presentation F3, Proceedings: 1992 EPRI Workshop on PWSCC of Alloy 600 in PWRs, EPRI TR-103345, December 1993.

24. Brown, C.M., and Mills, W.J., "Fracture Toughness, Tensile and Stress Corrosion Cracking Properties of Alloy 600, Alloy 690, and Their Welds in Water," paper 90, Corrosion96, NACE, 1996.

25. Mills, W.J., and C.M. Brown, "Fracture Behavior of Nickel-based Alloys in Water," Ninth International Conference on Environmental Degradation of Materials in Nuclear Power Systems--Water Reactors, August 1-5, 1999, Newport Beach, CA, TMS/NACE.

26. Mills, W.J., Brown, C.M., and Burke, M.G., "Effect of Microstructure on Low Temperature Cracking Behavior of EN82H Welds," Tenth International Conference on Environmental Degradation of Materials in Nuclear Power Systems--Water Reactors, August 2001, NACE.

27. Mills, W.J., Lebo, M.R., and Kearns, J.J., "Hydrogen Embrittlement, Grain Boundary Segregation and Stress Corrosion Cracking of Alloy X-750 in Low and High Temperature Water," Metallurgical and Materials Transactions, Vol. 30A, 1999, pp. 1579-1596.

28. Scott, P., "A Review of Irradiation Assisted Stress Corrosion Cracking," Journal of Nuclear Materials, Volume 211, 1994, pp. 101-122.

29. "Critical Issue Reviews for the Understanding and Evaluation of Irradiation-Assisted Stress Corrosion Cracking," EPRI TR-107159, November 1996.

30. Yukawa, S., "Review and Evaluation of the Toughness of Austenitic Steels and Nickel Alloys After Long-Term Elevated Temperature Exposures," WRC Bulletin 378, Welding Research Council, New York, January 1993.

31. Chopra, O.K., "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1991.

32. Xu, H. and Fyfitch, S., "Aging Embrittlement Modeling of Type 17-4 PH at LWR Temperatures," The 10th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, NACE International, Houston, Texas (2001).

33. Peckner, D., and Bernstein, I. M., Eds., Handbook of Stainless Steels, McGraw-Hill, New York, 1977.

34. Chopra, O.K. and Chung, H.M., "Aging Degradation of Cast Stainless Steels: Effects on Mechanical Properties," The 3rd International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, TMS 1987.

35. Clarke, Jr., W. C., "A Study of Embrittlement of a Precipitation Hardening Stainless Steel and Some Related Materials," Transactions of the Metallurgical Society of AIME, 245, 2135-2140, (October 1969).

36. "Materials Handbook for Nuclear Plant Pressure Boundary Applications," TR-109668-S1-R1, Electric Power Research Institute, December 1998.

37. "PWR Reactor Coolant System License Renewal Industry Report," Project RP-2643-32, Electric Power Research Institute, Palo Alto, California, May 1992.

38. NEI document: "Integrated Materials Issue Strategic Plan," Rev. 0, March 2004.

39. Keisler, J., Chopra, O.K., and Shack, W.J., "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," Argonne National Laboratory, ANL-95/15, NUREG/CR-6335, 1995.

40. Manjoine, M. J., and Voorhees, H. R., "Compilation of Stress-Relaxation Data for Engineering Alloys," ASTM Data Series, 1982, ASTM Philadelphia, PA.

41. Porollo, S.I., Vorobjev, A.N., Konobeev, Yu.V., Dvoriashin, A.M., Krigan, V.M., Budylkin, N.I., Mironova, E.G., and Garner, F.A., “Swelling and Void-Induced Embrittlement of Austenitic Stainless Steel Irradiated to 73-82 dpa at 335-365°C,” Journal of Nuclear Materials 258-263 (1998) pp. 1613-1617.

42. Porollo, S.I., Dvoriashin, A.M., Vorobjev, A.N., Krigan, V.M., Konobeev, Yu.V., Garner, F.A., Budylkin, N.I., and Mironova, E.G., “Neutron-Induced Swelling and Embrittlement Behavior of Two Russian Stainless Steels at PWR-Relevant Temperatures and 65-85 dpa,” Proc. Ninth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, The Minerals, Metals, and Materials Society (TMS), 1999, pp. 1060-1067.

43. Chin, B. A., Straalsund, J. L. and Wire, G. L., “Irradiation Creep of Previously Irradiated 304 Stainless Steel in a Fast Flux Environment,” Journal of Nuclear Materials, Vol. 75, 1978, pp. 274-281.

44. Garner, F. A. and Gelles, D. S., “Neutron-Induced Swelling of Commercial Alloys at Very High Exposures,” STP 1046, Effects of Radiation on Materials, Fourteenth International Symposium, ASTM, Philadelphia, PA, 1990.

45. EPRI-TR-103838s, “PWR Reactor Pressure Vessel Internals License Renewal Industry Report; Revision 1”, EPRI, Palo Alto, CA, July 1994.

46. License Renewal Evaluation: Aging Management for Reactor Internals, Westinghouse Electric Corp., WCAP-14577, Rev. 1-A, March 2001.

47. “Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems,” O. K. Chopra, NUREG/CR-4513 (ANL-93/22), Revision 1, Argonne National Laboratory, August 1994.

48. Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Vol. 1., U. S. Nuclear Regulatory Commission, July 2001.

49. Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, July 2001.

50. EPRI TR-107079, “BWR Vessel and Internals Project: BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-01),” EPRI, Palo Alto, CA, October 1996.

51. EPRI TR-114232, “BWR Vessel and Internals Project: BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-76),” EPRI, Palo Alto, CA, November 1999.

A
PWR INTERNALS MATERIALS OF CONSTRUCTION

Materials for the PWR internals were chosen to meet ASME B&PV Code requirements. A generic list of PWR internals materials of construction is given in Table A-1. This materials information is important for the screening and flaw tolerance evaluations to be performed when developing a plant-specific inspection plan for aging management of the PWR internals.

B
AGING MANAGEMENT PROGRAM ATTRIBUTES

Aging management programs are generally of four types: prevention, mitigation, condition monitoring, and performance monitoring. Prevention programs preclude the effects of aging. Mitigation programs attempt to slow the effects of aging. Condition monitoring programs (including surveillance sampling) detect the presence and extent of aging effects. Performance monitoring programs test the ability of a structure or component to continue to perform its intended function(s) in the presence of aging. An acceptable aging management program should consist of the 10 attributes described in Table B-1, as appropriate.[48] In the GALL Report, these program elements/attributes are discussed further, as described below:

1) Scope of Program

The specific program necessary for license renewal should be identified. The scope of the program should include the specific structures and components of which the program manages the aging.

2) Preventive Actions

The activities for prevention and mitigation programs should be described. These actions should mitigate or prevent aging degradation.

3) Parameters Monitored or Inspected

The parameters to be monitored or inspected should be identified and linked to the degradation of the particular structure and component intended function(s).

4) Detection of Aging Effects

Detection of aging effects should occur before there is a loss of the structure and component intended function(s). The parameters to be monitored or inspected should be appropriate to ensure that the structure and component intended function(s) will be adequately maintained for license renewal under all CLB design conditions. This includes aspects such as method or technique (e.g., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects. Information should be provided that links the parameters to be monitored or inspected to the aging effects being managed.

Nuclear power plants are licensed based on redundancy, diversity, and defense-in-depth principles. A degraded or failed component reduces the reliability of the system, challenges safety systems, and contributes to plant risk. Thus, the effects of aging on a structure or component should be managed to ensure its availability to perform its intended function(s) as designed when called upon. In this way, all system level intended function(s), including redundancy, diversity, and defense-in-depth consistent with the plant's CLB, would be maintained for license renewal. A program based solely on detecting structure and component failure should not be considered as an effective aging management program for license renewal.

This program element describes "when," "where," and "how" program data are collected (i.e., all aspects of activities to collect data as part of the program).

The method or technique and its frequency of application may be linked to plant-specific or industry-wide operating experience. Justification should be provided, including codes and standards referenced, that the technique and its frequency of application are adequate to detect the aging effects before a loss of intended function in structures and components (SC). A program based solely on detecting SC failures is not considered an effective aging management program.

When surveillance sampling is used to monitor a group of SCs, provide the basis for surveillance population and sample size. The inspection population should be based on such aspects of the SCs as a similarity of materials of construction, fabrication, procurement, design, installation, operating environment, or aging effects. The sample size should be based on such aspects of the SCs as the specific aging effect, location, existing technical information, system and structure design, materials of construction, service environment, or previous failure history. The samples should be biased toward locations most susceptible to the specific aging effect of concern in the period of extended operation. Provisions should also be included on expanding the sample size when degradation is detected in the initial sample.

5) Monitoring and Trending

Monitoring and trending activities should be described, and they should provide predictability of the extent of degradation and thus effect timely corrective or mitigative actions. Plant-specific and/or industry-wide operating experience may be considered in evaluating the appropriateness of the technique and frequency.

This program element describes "how" the data collected are evaluated and may also include trending for a forward look. This includes an evaluation of the results against the acceptance criteria and a prediction regarding the rate of degradation in order to confirm that timing of the next scheduled inspection will occur before a loss of SC intended function. Although aging indicators may be quantitative or qualitative, aging indicators should be quantified, to the extent possible, to allow trending. The parameter or indicator trended should be described. The methodology for analyzing the inspection or test results against the acceptance criteria should be described. Trending is a comparison of the current monitoring results with previous monitoring results in order to make predictions for the future.

6) Acceptance Criteria

The acceptance criteria of the program and its basis should be described. The acceptance criteria, against which the need for corrective actions will be evaluated, should ensure that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation. The program should include a methodology for analyzing the results against applicable acceptance criteria.

Acceptance criteria could be specific numerical values, or could consist of a discussion of the process for calculating specific numerical values of conditional acceptance criteria to ensure that the structure and component intended function(s) will be maintained under all CLB design conditions. Information from available references may be cited.

It is not necessary to justify any acceptance criteria taken directly from the design basis information that is included in the FSAR because that is a part of the CLB. Also, it is not necessary to discuss CLB design loads if the acceptance criteria do not permit degradation because a structure and component without degradation should continue to function as originally designed. Acceptance criteria, which do permit degradation, are based on maintaining the intended function under all CLB design loads.

Qualitative inspections should be performed to same predetermined criteria as quantitative inspections by personnel in accordance with ASME B&PV Code and through approved site specific programs.

7) Corrective Actions

Actions to be taken when the acceptance criteria are not met should be described. Corrective actions, including root cause determination and prevention of recurrence, should be timely.

If corrective actions permit analysis without repair or replacement, the analysis should ensure that the structure and component intended function(s) will be maintained consistent with the CLB.

8) Confirmation Process

The confirmation process should be described. It should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.

The effectiveness of prevention and mitigation programs should be verified periodically. For example, in managing internal corrosion of piping, a mitigation program (water chemistry) may be used to minimize susceptibility to corrosion. However, it may also be necessary to have a condition monitoring program (ultrasonic inspection) to verify that corrosion is indeed insignificant.

When corrective actions are necessary, there should be follow-up activities to confirm that the corrective actions were completed, the root cause determination was performed, and recurrence is prevented.

9) Administrative Controls

The administrative controls of the program should be described. They should provide a formal review and approval process.

Any aging management programs to be relied on for license renewal should have regulatory and administrative controls. That is the basis for 10 CFR 54.21(d) to require that the FSAR supplement includes a summary description of the programs and activities for managing the effects of aging for license renewal. Thus, any informal programs relied on to manage aging for license renewal must be administratively controlled and included in the FSAR supplement.

10) Operating experience

Operating experience with existing programs should be discussed. The operating experience of aging management programs, including past corrective actions resulting in program enhancements or additional programs, should be considered. A past failure would not necessarily invalidate an aging management program because the feedback from operating experience should have resulted in appropriate program enhancements or new programs. This information can show where an existing program has succeeded and where it has failed (if at all) in intercepting aging degradation in a timely manner. This information should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

An applicant may have to commit to providing operating experience in the future for new programs to confirm their effectiveness.

Table B-1
Elements of an Aging Management Program for License Renewal [48]

Element	Description
1. Scope of program	Scope of program should include the specific structures and components subject to an AMR for license renewal.
2. Preventive actions	Preventive actions should prevent or mitigate aging degradation.
3. Parameters monitored or inspected	Parameters monitored or inspected should be linked to the degradation of the particular structure or component intended function(s).
4. Detection of aging effects	Detection of aging effects should occur before there is a loss of structure or component intended function(s). This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and trending	Monitoring and trending should provide predictability of the extent of degradation, and timely corrective or mitigative actions.
6. Acceptance criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation.
7. Corrective actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely.
8. Confirmation process	Confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
9. Administrative controls	Administrative controls should provide a formal review and approval process.
10. Operating experience	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

C
ACRONYMS AND GLOSSARY OF TERMS

ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CASS	Cast Austenitic Stainless Steel
C-E	Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CRA	Control Rod Assemblies
DH	Dissolved Hydrogen
DM	Degradation Matrix
ECP	Electrochemical Potential
EFPY	Effective Full Power Years
EPRI	Electric Power Research Institute
FIV	Flow-Induced Vibration
GALL	Generic Aging Lessons Learned (Report)
HC	High-Cycle (Fatigue)
IASCC	Irradiation-Assisted Stress Corrosion Cracking

IC	Irradiation Creep
IGSCC	Intergranular Stress Corrosion Cracking
IE	Irradiation Embrittlement
IMT	Issue Management Table
ISI	In-Service Inspection
JOBB	Joint Owners Baffle Bolt Program
LC	Low-Cycle (Fatigue)
LTCP	Low Temperature Crack Propagation
MRP	Material Reliability Program
MTAG	Materials Technical Advisory Group
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PH	Precipitation-Hardenable
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCCA	Rod Cluster Control Assemblies
RI-ITG	Reactor Internals Issues Task Group
RIS	Radiation-Induced Segregation
SCC	Stress Corrosion Cracking
SR	Stress Relaxation
TA	Thermal Aging
TGSCC	Transgranular Stress Corrosion Cracking
VS	Void Swelling

W	Westinghouse Electric Corporation
WCAP	Westinghouse Commercial Atomic Power (Report)

Acceptance By Visual Examination	A component whose visual examination confirms the absence of the relevant conditions described in the standards of Table IWB-3410-1 shall be acceptable for service (IWB-3122.1(a)). A component whose visual examination detects the relevant conditions described in the standards of Table-3410-1 shall be unacceptable for service, unless such components meet the requirements of IWB-3122.2 or IWB-3122.3 prior to placement of the component in service (IWB-3122.1(b)).
Character Recognition Demonstration	The demonstration that VT-1 and VT-3 examination techniques are capable of representative lower case characters of dimensions, at distances from, and under illumination conditions specified in Table IWA-2211-1. For VT-1 examination, the specified character height is 0.044 in. (1.1 mm) and the maximum standoff distance is 24 in. (610 mm). For VT-3 examination, the specified character height is 0.105 in. (2.7 mm) and the maximum standoff distance is 72 in. (1219 mm) (adapted from IWA-2210, ASME B&PV Code Section XI).
Core Support Structures	Those structures or parts of structures that are designed to provide direct support or restraint of the core (fuel and blanket assemblies) within the reactor pressure vessel (IWA-9000).
Crack Growth in Austenitic Components	The stable flaw extension caused by cyclic fatigue flaw growth, stress corrosion cracking under sustained load, or a combination of both (adapted from C-3200, Appendix C, ASME B&PV Code Section XI).
Crack Initiation	The onset of flaw extension due to an increase in component loading (Appendix A, ASME B&PV Code Section XI).
Crack Tightness	The characteristic magnitude of the crack opening surface displacement of a surface-breaking crack following removal of the applied load causing crack propagation (new definition).
Damage Tolerant	This term refers to components that can continue to perform their intended functions in a degraded condition. The term is a generalization of the concept of flaw tolerance (new definition).

Defect	A flaw (imperfection or unintentional discontinuity) of such size, shape, orientation, location, or properties as to be rejectable (IWA-9000). One or more flaws whose aggregate size, shape, orientation, location, or properties do not meet specified acceptance criteria and are rejectable (Article 30, ASME B&PV Code Section V).
Direct Visual Examination	A visual examination technique performed by eye and without any visual aids (excluding light source, mirrors, and/or corrective lenses (Article 9, ASME B&PV Code Section V).
Discontinuity	A lack of continuity or cohesion; an interruption in the normal physical structure of material or a product (IWA-9000).
Eddy Current Testing	A nondestructive test method in which eddy current flow is induced in the test object. Changes in the flow caused by variations in the specimen are reflected into a nearby coil, coils, or Hall effect device for subsequent analysis by suitable instrumentation and techniques (Article 30, ASME B&PV Code Section V).
Enhanced Visual Examination	A visual examination technique using visual aids to improve the viewing capability, e.g., magnifying aids, borescopes, video probes, fiber optics, etc. (Article 9, ASME B&PV Code Section V).
Enhanced VT-1 Visual Examination	Enhanced visual examination (EVT-1), as defined in BWRVIP-03,[6] is a visual examination method where the equipment and the environmental conditions are such that the detection of a 1/2 mil (0.0005 inches or 0.0127 mm) wire can be demonstrated.
Examination Category	A grouping of items to be examined or tested (IWA-9000).
Examination Category B-N-1	The examination category that includes accessible areas of the reactor vessel interior using VT-3 visual examination techniques (adapted from Table IWB-2500-1).
Examination Category B-N-2	The examination category that includes accessible welds for interior attachments within the reactor vessel beltline using VT-1 visual examination techniques, and accessible welds for interior attachments outside the reactor vessel beltline using VT-3 visual examination techniques (adapted from Table IWB-2500-1).
Examination Category B-N-3	The examination category that includes accessible (or made accessible by removal) surfaces of core support structures using VT-3 visual examination techniques (adapted from IWB-2500-1).

Flaw	An imperfection or discontinuity that may be detectable by nondestructive testing and is not necessarily rejectable (Article 30, ASME B&PV Code Section V).
Functionality Assessment	A quantitative assessment of the consequences of significant age-related degradation effects on the functional performance of a component. The goal of a functionality assessment is to determine whether or not a component is damage tolerant.
Indication	The response or evidence from the application of a nondestructive examination (IWA-9000, ASME B&PV Code Section XI).
Inservice Examination	The process of visual, surface, or volumetric examination performed in accordance with the rules and requirements of Division 1 of the ASME B&PV Code Section XI (adapted from IWA-9000).
Inservice Inspection	Methods and actions for assuring the structural and pressure-retaining integrity of safety-related nuclear power plant components in accordance with the rules of the ASME B&PV Code Section XI (adapted from IWA-9000).
Lead Component	A PWR internals component for which aging effects are above screening levels, which have moderate or high susceptibility to degradation, and have not been demonstrated to be sufficiently damage tolerant to remain functional relative to aging degradation significance.
Linear Elastic Fracture Mechanics	The analytical procedure that relates the stress-field magnitude and distribution in the vicinity of a crack tip, resulting from the nominal stress applied to the structure, to the size of a crack that would cause non-ductile failure (Appendix A, ASME B&PV Code Section XI).
Liquid Penetrant Examination	A nondestructive test that uses suitable liquids that penetrate discontinuities open to the surface of solid materials and, after appropriate treatment, indicate the presence of discontinuities (Article 30, ASME B&PV Code Section V).
Magnetic Particle Examination	A nondestructive test method utilizing magnetic leakage fields and suitable indicating materials to disclose surface and near-surface discontinuity indications (Article 30, ASME B&PV Code Section V).

Mature Crack	A surface-breaking crack propagated to a depth under applied load such that the crack opening surface displacement is of the same order of magnitude as the character recognition height demonstration requirements of Table IWA-2211-1 of the ASME B&PV Code Section XI (new definition).
Nondestructive Examination	An examination by the visual, surface, or volumetric method (IWA-9000). The development and application of technical methods to examine materials and/or components in ways that do not impair future usefulness and serviceability in order to detect, locate, measure, interpret, and evaluate flaws (Article 1, I-130, ASME B&PV Code Section V).
Radiographic Inspection	The use of X-rays or nuclear radiation, or both, to detect discontinuities in material, and to present their images on a recording medium (Article 30, ASME B&PV Code Section V).
Relevant Condition	A condition observed during a visual examination that requires supplemental examination, corrective measure, correction by repair/replacement activities, or analytical evaluation (IWA-9000, ASME B&PV Code Section XI).
Relevant Indication	An indication detected by nondestructive testing that is caused by a condition or type of discontinuity that requires evaluation (Adapted from Article 30, ASME B&PV Code Section V).
Remote Visual Examination	A visual examination technique used with visual aids for conditions where the area to be examined is inaccessible for direct visual examination (Article 9, ASME B&PV Code Section V).
Supplemental Examination	A surface or volumetric examination to determine the extent of the unacceptable conditions and the need for corrective measures, analytical evaluation or repair/replacement activities, based on the detection of relevant conditions by visual examination (adapted from IWB-3200, ASME B&PV Code Section XI).
Ultrasonic Testing (UT)	A nondestructive method of examining materials by introducing ultrasonic waves into, through, or onto the surface of the article being examined and determining various attributes of the material from effects on the ultrasonic waves (Adapted from Article 30, ASME B&PV Code Section V).

Visual (VT) Examination	A nondestructive examination method used to evaluate an item by observation, such as: the correct assembly, surface conditions, or cleanliness of materials, parts, and components used in the fabrication and construction of ASME B&PV Code vessels and hardware (Article 9, ASME B&PV Code Section V).
VT-1 Visual Examination	A visual examination technique conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion, in accordance with the requirements of Table IWA-2211-1 (adapted from IWA-2211).
VT-3 Visual Examination	A visual examination technique conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements; and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion, in accordance with the requirements of Table IWA-2211-1 (adapted from IWA-2213, ASME B&PV Code Section XI).
VT-3 Visual Examination Standards	The following relevant conditions shall require correction in meeting the requirements of IWB-3122 prior to service or IWB-3142 prior to continued service: (a) structural distortion or displacement of parts to the extent that component function may be impaired; (b) loose, missing, cracked, or fractured parts, bolting, or fasteners; (c) foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel; (d) corrosion or erosion that reduces the nominal section thickness by more than 5 %; (e) wear of mating surfaces that may lead to loss of function; and structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 % (IWB-3520.2).