CHAPTER 3

AGING MANAGEMENT REVIEW RESULTS

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3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Review Responsibilities

Primary - Branch responsible for materials and chemical engineering
Secondary - Branch responsible for mechanical engineering

3.1.1 Areas of Review

This review plan section addresses the aging management review (AMR) of the reactor vessel, internals, and reactor coolant system. For a recent vintage plant, the information related to the reactor vessel, internals, and reactor coolant system is contained in Chapter 5, "Reactor Coolant System and Connected Systems," of the plant's final safety analysis report (FSAR), consistent with the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800) (Ref. 1). For older plants, the location of applicable information is plant-specific because their FSAR may have predated NUREG-0800.

The reactor vessel, internals, and reactor coolant system includes the reactor vessel and internals. Also included for BWRs are the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the first isolation valve outside of containment or to the first anchor point. These connected systems include residual heat removal, low-pressure core spray, high-pressure core spray, low-pressure coolant injection, high-pressure coolant injection, reactor core isolation cooling, isolation condenser, reactor coolant cleanup, feedwater, and main steam. For PWRs, the reactor coolant system includes the primary coolant loop, the pressurizer and pressurizer relief tank, and the steam generators. The connected systems for PWRs include the residual heat removal or low pressure injection system, core flood spray or safety injection tank, chemical and volume control system or high pressure injection system, and sampling system.

The staff has issued a generic aging lessons learned (GALL) report addressing aging management for license renewal (Ref. 2). The GALL report documents the staff's basis for determining whether generic existing programs are adequate to manage aging without change or generic existing programs should be augmented for license renewal. The GALL report may be referenced in a license renewal application and should be treated in the same manner as an approved topical report.

Because a license renewal applicant may or may not be able to reference the GALL report as explained below, the following areas are reviewed.

3.1.1.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

The applicant may reference the GALL report in a license renewal application to demonstrate that the programs at its facility correspond to those reviewed and approved in the report and that no further staff review is required. If the material presented in the GALL report is applicable to the applicant's facility, the staff should find the applicant's reference to the report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL report. The staff, however, should not repeat its review of the substance of the matters described in the report. Rather, the staff should confirm that the applicant verifies that the approvals set forth in the GALL report for generic programs apply to the applicant's programs.

3.1.1.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report provides the basis for identifying those programs that warrant further evaluation during the staff review of a license renewal application. The staff review focus should be on augmented programs for license renewal.

3.1.1.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

The GALL report provides a generic staff evaluation of certain aging management programs (AMPs). If the applicant does not rely on a particular program for license renewal, or if the applicant indicates that the generic staff evaluation of the elements of a particular program does not apply to its plant, the staff should review each such AMP to which the GALL report does not apply.

The GALL report provides a generic staff evaluation of programs for certain components and aging effects. If the applicant has identified particular components subject to aging management review (AMR) for its plant that are not addressed in the GALL report, or particular aging effects for a component that are not addressed in the GALL report, the staff should review the applicant's AMPs applicable to these particular components and aging effects.

3.1.1.4 FSAR Supplement

The FSAR supplement summarizing the programs and activities for managing the effects of aging for the period of extended operation is reviewed.

3.1.2 Acceptance Criteria

The acceptance criteria for the areas of review describe methods for determining whether the applicant has met the requirements of the NRC's regulations in 10 CFR 54.21.

3.1.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Acceptable methods for managing aging of the reactor vessel, internals, and reactor coolant system are described and evaluated in Chapter IV of the GALL report (Ref. 2). In referencing this report, the applicant should indicate that the material presented is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for generic programs apply to the applicant's programs. The applicant may reference appropriate programs as described and evaluated in the GALL report.

3.1.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report indicates that further evaluation should be performed for the following.

3.1.2.2.1 Cumulative Fatigue Damage (BWR/PWR)

Fatigue is a time-limited aging analysis (TLAA) as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.3.

3.1.2.2.2 Loss of Material due to Pitting and Crevice Corrosion (BWR/PWR)

1. Loss of material due to pitting and crevice corrosion could occur in the PWR steam generator shell assembly. The existing program relies on control of chemistry to mitigate corrosion and ISI to detect loss of material. The extent and schedule of the existing steam generator inspections are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, according to NRC Information Notice (IN) 90-04 (Ref. 4), if general corrosion pitting of the shell exists, the program may not be sufficient to detect pitting and corrosion. The GALL report recommends augmented inspection to manage this aging effect. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Loss of material due to pitting and crevice corrosion could occur in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ASME Section XI inservice inspection (ISI). However, the existing program should be augmented to detect loss of material due to pitting or crevice corrosion. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes to ensure that the component's intended function will be maintained during the period of extended operation. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement (BWR/PWR)

1. Certain aspects of neutron irradiation embrittlement are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.2.

2. Loss of fracture toughness due to neutron irradiation embrittlement could occur in the reactor vessel. A reactor vessel materials surveillance program monitors neutron irradiation embrittlement of the reactor vessel. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant is required to submit its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur in Westinghouse and B&W baffle/former bolts. The GALL report recommends further

evaluation to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.4 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR/PWR)

1. Crack initiation and growth due to thermal and mechanical loading or SCC (including intergranular stress corrosion cracking [IGSCC]) could occur in small-bore reactor coolant system and connected system piping less than NPS 4. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate SCC. The GALL report recommends that a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. The AMPs should be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. A one-time inspection of a sample of locations is an acceptable method to ensure that the aging effect is not occurring and the component's intended function will be maintained during the period of extended operation.

2. Crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) could occur in BWR reactor vessel flange leak detection line and BWR jet pump sensing line. The GALL report recommends that a plant specific aging management program be evaluated to mitigate or detect crack initiation and growth due to SCC of vessel flange leak detection line. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) could occur in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate SCC and on ASME Section XI inservice inspection (ISI). However, the existing program should be augmented to detect cracking due to SCC or cyclic loading. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes to ensure that the component's intended function will be maintained during the period of extended operation.

3.1.2.2.5 Crack Growth due to Cyclic Loading (PWR)

Crack growth due cyclic loading could occur in reactor vessel shell and reactor coolant system piping and fittings. Growth of intergranular separations (underclad cracks) in low-alloy or carbon steel heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. See the Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," for generic guidance for meeting the requirements of 10 CFR 54.21(c).

3.1.2.2.6 Changes in Dimension due to Void Swelling (PWR)

Changes in dimension due to void swelling could occur in reactor internal components. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. GALL recommends that a plant-specific aging management program should be evaluated. The applicant provides a plant-specific AMP or participates in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant provides the basis for concluding that void swelling is not an issue for the component. The applicant should either provide the basis for concluding that void swelling is not an issue for the component or provide a program to manage the effects of changes in dimension due to void swelling and the loss of ductility associated with swelling.

3.1.2.2.7 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (PWR)

1. Crack initiation and growth due to SCC and primary water stress corrosion cracking (PWSCC) could occur in PWR core support pads (or core guide lugs), instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains. The GALL report recommends further evaluation to ensure that these aging effects are adequately managed. The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Crack initiation and growth due to SCC could occur in PWR cast austenitic stainless steel (CASS) reactor coolant system piping and fittings and pressurizer surge line nozzle. The GALL report recommends further evaluation of piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313 (Ref. 5). Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Crack initiation and growth due to PWSCC could occur in PWR pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni alloys. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate PWSCC. However, the existing program should be augmented to manage the effects of SCC on the intended function of Ni-alloy components. The GALL report recommends that the applicant provide a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.8 Crack Initiation and Growth due to Stress Corrosion Cracking or Irradiation-Assisted Stress Corrosion Cracking (PWR)

Crack initiation and growth due to SCC or IASCC could occur in baffle/former bolts in Westinghouse and B&W reactors. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts at several plants have identified cracking. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis. The GALL report recommends further evaluation to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.9 Loss of Preload due to Stress Relaxation (PWR)

Loss of preload due to stress relaxation could occur in baffle/former bolts in Westinghouse and B&W reactors. Visual inspection (VT–3) should be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is thus required. The GALL report recommends a plant-specific aging management program to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.10 Loss of Section Thickness due to Erosion (PWR)

Loss of section thickness due to erosion could occur in steam generator feedwater impingement plates and supports. The GALL report recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.11 Crack Initiation and Growth due to PWSCC, ODSCC, or Intergranular Attack or Loss of Material due to Wastage and Pitting Corrosion or Loss of Section Thickness due to Fretting and Wear or Denting due to Corrosion of Carbon Steel Tube Support Plate (PWR)

Crack initiation and growth due to PWSCC, ODSCC, or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion or deformation due to corrosion could occur in alloy 600 components of the steam generator tubes, repair sleeves and plugs. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, should be developed to ensure that this aging effect is adequately managed.

3.1.2.2.12 Loss of Section Thickness due to Flow-accelerated Corrosion

Loss of section thickness due to flow-accelerated corrosion could occur in tube support lattice bars made of carbon steel. The GALL report recommends that a plant-specific aging management program be evaluated and, on the basis of the guidelines of NRC Generic Letter 97-06, an inspection program for steam generator internals be developed to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.13 Ligament Cracking due to Corrosion (PWR)

Ligament cracking due to corrosion could occur in carbon steel components in the steam generator tube support plate. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed.

3.1.2.2.14 Loss of Material due to Flow-accelerated Corrosion (PWR)

Loss of material due to flow-accelerated corrosion could occur in feedwater inlet ring and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting loss of material due to flow-accelerated corrosion. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.15 Quality Assurance for Aging Management of Nonsafety-Related Components

Acceptance criteria are described in Branch Technical Position IQMB-1 (Appendix A.2 of this standard review plan).

3.1.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.4 FSAR Supplement

The summary description of the programs and activities for managing the effects of aging for the period of extended operation in the FSAR supplement should be appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the bases for determining that aging effects will be managed during the period of extended operation.

3.1.3 Review Procedures

For each area of review, the following review procedures are to be followed.

3.1.3.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

The applicant may reference the GALL report in its license renewal application, as appropriate. The staff should not repeat its review of the substance of the matters described in the report. If the applicant has provided the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report, the staff should find the applicant's reference to the report in a license renewal application acceptable. In making this determination, the reviewer verifies that the applicant has provided a brief description of the system, components, materials, and environment. The reviewer also verifies that the applicant has stated that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL report. The reviewer verifies that the applicant has identified those aging effects for the reactor vessel, internals, and reactor coolant system components that are contained in the report as applicable to its plant. In addition, the reviewer ensures that the applicant has stated that the plant programs covered by the applicant's reference contain the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report.

The reviewer should verify that the applicant has stated that certain of its AMPs contain the same program elements as the corresponding generic program described in the GALL report and upon which the staff relied in its evaluation. The reviewer should also verify that the applicant has stated that the GALL report is applicable to its plant with respect to these programs. The reviewer verifies that the applicant has identified the appropriate programs as described and evaluated in the GALL report. Programs evaluated in the report regarding the reactor vessel, internals, and reactor coolant system components are summarized in Table 3.1-1 of this review plan. No further staff evaluation is necessary if so recommended in the GALL report.

3.1.3.2 Further Evaluation of Aging Management as Recommended by the GALL Report

3.1.3.2.1 Cumulative Fatigue Damage (BWR/PWR)

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.3 of this standard review plan.

3.1.3.2.2 Loss of Material due to Pitting and Crevice Corrosion (BWR/PWR)

1. The GALL report recommends further evaluation for the management of loss of material due to pitting and crevice corrosion of the PWR steam generator shell assembly. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ISI for detection. Based on NRC IN 90-04 (Ref. 4), if general corrosion pitting of the shell exists, the existing program requirements may not be sufficient to detect loss of material due to pitting and corrosion, and additional inspection procedures may be required. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage loss of material due to pitting and crevice corrosion by providing enhanced inspection and supplemental methods to detect loss of material and ensure that the component intended function will be maintained during the extended period.

2. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water and eddy current testing of tubes for the management of loss of material due to pitting and crevice corrosion in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ASME Section XI ISI for detection. However, the inspection requirements should be augmented to detect loss of material due to pitting and crevice corrosion, and an augmented program to include temperature and radioactivity monitoring of the shell-side water and eddy current testing of tubes is recommended to ensure that the component's intended function will be maintained during the period of extended operation. The reviewer verifies on a case-by-case basis that the applicant has proposed an augmented program that will manage loss of material due to pitting and crevice corrosion and ensure that the component intended function will be maintained during the extended period.

3.1.3.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement (BWR/PWR)

1. Neutron irradiation embrittlement is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA following the guidance in Section 4.2 of this standard review plan.

2. The GALL report recommends further evaluation of the reactor vessel materials surveillance program for the period of extended operation. Neutron embrittlement of the reactor vessel is monitored by a reactor vessel materials surveillance program. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant must submit its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal. The reviewer verifies on a case-by-case basis that the applicant has proposed an adequate reactor vessel materials surveillance program for the period of extended operation. Specific criteria for an acceptable AMP is provided in chapter XI, Section M31 of the GALL report.

3. The GALL report recommends further evaluation for the management of loss of fracture toughness due to neutron irradiation embrittlement and void swelling of Westinghouse and B&W baffle/former bolts. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.4 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR/PWR)

1. The GALL report recommends a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping for the management of crack initiation and growth due to thermal and mechanical loading or SCC of small-bore reactor coolant system and connected system piping (less than NPS 4). The existing program should be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method. The GALL report recommends that the inspection include a representative sample of the system population, and, where practical and prudent, focus on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. For small-bore piping, actual inspection locations should be based on physical accessibility, exposure levels, NDE examination techniques, and locations identified in Nuclear Regulatory Commission (NRC) Information Notice (IN) 97-46. Combinations of NDE, including visual, ultrasonic, and surface techniques, are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50 Appendix B. For small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, a plant-specific destructive examination or NDE that permits inspection of the inside surfaces of the piping should be conducted to ensure that cracking has not occurred. Follow-up of unacceptable inspection findings should include expansion of the inspection sample size and locations. The inspection and test techniques prescribed by the program should verify any aging effects because these techniques, used by qualified personnel, have been proven effective and consistent with staff expectations. The staff reviews to confirm that the program includes measures to verify that unacceptable

degradation is not occurring, thereby validating the effectiveness of existing programs or confirming that there is no need to manage aging-related degradation for the period of extended operation. If an applicant proposes a one-time inspection of select components and susceptible locations to ensure that corrosion is not occurring, the reviewer verifies that the proposed inspection will be performed using techniques similar to ASME Code and ASTM standards including visual, ultrasonic, and surface techniques (Refs. 6 and 7) to ensure that the component's intended function will be maintained during the period of extended operation.

2. The GALL report recommends that a plant specific aging management program be evaluated for the management of crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) in BWR reactor vessel flange leak detection line and BWR jet pump sensing line. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes for the management of crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) of the BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate SCC and on ASME Section XI inservice inspection (ISI) to detect leakage. However, the existing program should be augmented to detect cracking due to SCC or cyclic loading. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.5 Crack Growth due to Cyclic Loading (PWR)

The GALL report recommends further evaluation of programs to manage crack growth due to cyclic loading in reactor vessel shell and reactor coolant system piping and fittings. Growth of intergranular separations (underclad cracks) in low-alloy or carbon steel heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. The Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," provides generic guidance for meeting the requirements of 10 CFR 54.21(c). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.7 of this standard review plan.

3.1.3.2.6 Changes in Dimension due to Void Swelling (PWR)

The GALL report recommends further evaluation of programs to manage changes in dimension due to void swelling for reactor internal components. Changes in dimension due to void swelling could occur in reactor internal components. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IWB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. The GALL report recommends further evaluation of a plant-specific aging management program. The applicant should provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine an appropriate AMP. Otherwise, the applicant should provide the basis for concluding that void swelling is not an issue for the component. The applicant should either provide the basis for concluding that void swelling is not an issue for the component or provide a program to manage the effects of changes in dimension due to void swelling and the loss of ductility associated with swelling. The reviewer verifies on a case-by-case basis that the applicant has either proposed a program to manage changes in dimension due to void swelling in the pressure vessel internal components or provided the basis for concluding that void swelling is not an issue.

3.1.3.2.7 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (PWR)

1. The GALL report recommends that a plant-specific aging management program is to be evaluated to manage crack initiation and growth due to SCC and primary water stress corrosion cracking (PWSCC) in PWR core support pads (or core guide lugs, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

2. The GALL report recommends further evaluation of programs to manage crack initiation and growth due to SCC of PWR cast austenitic stainless steel (CASS) reactor coolant system piping and fittings and pressurizer surge line nozzle. The GALL report recommends further evaluation of piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313 (Ref. 5). The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3. The GALL report recommends further evaluation of programs to manage crack initiation and growth due to PWSCC of PWR pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni alloys. The existing program relies on ASME Section XI ISI to detect cracks and on control of water chemistry to mitigate PWSCC. However, the program should be augmented to manage the effects of SCC on the intended function of Ni-alloy components. The GALL report recommends the applicant provides a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.8 Crack Initiation and Growth due to Stress Corrosion Cracking or Irradiation-Assisted Stress Corrosion Cracking (PWR)

The GALL report recommends further evaluation of crack initiation and growth due to SCC or IASCC in Westinghouse and B&W baffle/former bolts. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts at several plants have identified cracking. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project ITG activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.9 Loss of Preload due to Stress Relaxation (PWR)

The GALL report recommends further evaluation of loss of preload due to stress relaxation could occur in baffle/former bolts in Westinghouse and B&W reactors. Visual inspection (VT–3) should be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is thus required. The GALL report recommends a plant-specific aging management program to ensure that these aging effects are adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.10 Loss of Section Thickness due to Erosion (PWR)

The GALL report recommends further evaluation of a plant-specific aging management program for the management of loss of section thickness due to erosion of steam generator feedwater impingement plates and supports. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.11 Crack Initiation and Growth due to PWSCC, ODSCC, or Intergranular Attack or Loss of Material due to Wastage and Pitting Corrosion or Loss of Section Thickness due to Fretting and Wear or Denting due to Corrosion of Carbon Steel Tube Support Plate (PWR)

The GALL report recommends further evaluation of (1) crack initiation and growth due to PWSCC, ODSCC, or intergranular attack (IGA); or (2) loss of material due to wastage and pitting corrosion; or (3) deformation due to corrosion in alloy 600 components of the steam generator tubes, repair sleeves, and plugs. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.12 Loss of Section Thickness due to Flow-accelerated Corrosion

The GALL report recommends further evaluation of loss of section thickness due to flow-accelerated corrosion of the tube support lattice bars made of carbon steel. The GALL report recommends a plant-specific aging management program be evaluated and on the basis of the guidelines of NRC Generic Letter 97-06, an inspection program for steam generator internals should be developed to ensure that this aging effect is adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.13 Ligament Cracking due to Corrosion (PWR)

The GALL report recommends further evaluation of ligament cracking due to corrosion in carbon steel components in the steam generator tube support plate. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.14 Loss of material due to Flow-accelerated Corrosion (PWR)

The GALL report recommends that a plant-specific aging management program be evaluated to manage loss of material due to flow-accelerated corrosion in the feedwater inlet ring and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.15 Quality Assurance for Aging Management of Nonsafety-Related Components

The applicant's aging management programs for license renewal should contain the elements of corrective actions, the confirmation process, and administrative controls. Safety-related components are covered by 10 CFR Part 50 Appendix B, which is adequate to address these program elements. However, Appendix B does not apply to nonsafety-related components that are subject to an AMR for license renewal. Nevertheless, the applicant has the option to expand the scope of its 10 CFR Part 50 Appendix B program to include these components and address the associated program elements. If the applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If the applicant chooses alternative means, the branch responsible for quality assurance should be requested to review the applicant's proposal on a case-by-case basis.

3.1.3.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Review procedures are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.3.4 FSAR Supplement

The reviewer verifies that the applicant has provided information, equivalent to that in Table 3.1-2, in the FSAR supplement for aging management of the reactor vessel, internals, and reactor coolant system for license renewal. The reviewer also verifies that the applicant has provided information, equivalent to that in Table 3.1-2, in the FSAR supplement for Subsection 3.1.3.3, "Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report."

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license conditions, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 3.1-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.1.4 Evaluation Findings

The reviewer verifies that the applicant has provided sufficient information to satisfy the provisions of this review plan section, and the staff's evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has demonstrated that the aging effects associated with the reactor vessel, internals, and reactor coolant system will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the current licensing basis during the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the programs and activities for managing the effects of aging for the reactor vessel, internals, and reactor coolant system as reflected in the license conditions.

3.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

3.1.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

2. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, April 2001.

3. NEI 97-06, "Steam Generator Program Guidelines," Nuclear Energy Institute, December 1997.

4. NRC Information Notice 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," U.S. Nuclear Regulatory Commission, January 26, 1990.

5. NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BRW Coolant Pressure Boundary Piping, U.S. Nuclear Regulatory Commission, January 1988.

6. EPRI TR-107569-V1R5, "PWR Steam Generator Examination Guidelines, Rev. 5," Electric Power Research Institute September 1997.

7. NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," U.S. Nuclear Regulatory Commission, June 1974.

8. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (for Comment)," U.S. Nuclear Regulatory Commission, May 1976.

9. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," U.S. Nuclear Regulatory Commission, August 3, 1995.

10. NRC Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," U.S. Nuclear Regulatory Commission, February 23, 1990.

11. NRC Information Notice 90-30, "Ultrasonic Inspection Techniques for Dissimilar Metal Welds," U.S. Nuclear Regulatory Commission, May 1, 1990.

12. NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," May 2, 1989.

13. NSAC-202L-R2, "Recommendations for an Effective Flow-accelerated Corrosion Program," Electric Power Research Institute.

14. NRC Information Notice 96-11, "Ingress of Demineralizer Resins Increase Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.

15. NRC Generic Letter 97-06, "Degradation of Steam Generator Internals," U.S. Nuclear Regulatory Commission, December 30, 1997.

16. BWRVIP-29 (EPRI TR-103515), *BWR Water Chemistry Guidelines-Revision 3, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

17. EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, Volumes 1 and 2, Electric Power Research Institute, Palo Alto, CA, April 1988.

18. EPRI TR-105714, *PWR primary Water Chemistry Guidelines-Revision 3*, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

19. NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping*, January 25, 1988.

20. NRC Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, April 1,1997.

21. NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, July 9, 1997.

22. NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials*.

23. NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

24. NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, Richard E. Johnson, U.S. Nuclear Regulatory Commission, June 1990.

25. EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, Palo Alto, CA, December 1995.

26. NEI letter dated Dec. 11, 1998, Dave Modeen to Gus Lainas, "Responses to NRC Requests for Additional Information (RAIs) on GL 97-01."

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/PWR	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (See Subsection 3.1.2.2.1)
PWR	Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated (See Subsection 3.1.2.2.2.1)
BWR	Isolation condenser	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	Yes, plant specific (See Subsection 3.1.2.2.2.2)
BWR/PWR	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA (See Subsection 3.1.2.2.3.1)
BWR/PWR	Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific (See Subsection 3.1.2.2.3.2)
PWR	Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.3.3)
BWR/PWR	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/inspected and detection of aging effects are to be further evaluated (See Subsection 3.1.2.2.4.1)
BWR	Jet pump sensing line, and reactor vessel flange leak detection line	Crack initiation and growth due to SCC, intergranular stress corrosion cracking (IGSCC), or cyclic loading	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.4.2)
BWR	Isolation condenser	Crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading;	Inservice inspection; water chemistry	Yes, plant specific (See Subsection 3.1.2.2.4.3)
PWR	Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA (See Subsection 3.1.2.2.5)

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.6)
PWR	PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.7.1)
PWR	Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific(See Subsection 3.1.2.2.7.2)
PWR	Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated (See Subsection 3.1.2.2.7.3)
PWR	Westinghouse and B&W baffle former bolts	Crack initiation and growth due to SCC and IASCC	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.8)
PWR	Westinghouse and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.9)
PWR	Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.10)

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	(Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion, and fretting and wear; or deformation due to corrosion at tube support plate intersections	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated (See Subsection 3.1.2.2.11)
PWR	Tube support lattice bars made of carbon steel	Loss of section thickness due to FAC	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.12)
PWR	Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated (See Subsection 3.1.2.2.13)
PWR (CE)	Steam generator feedwater inlet ring and supports	Loss of material due to flow-corrosion	Combustion engineering (CE) steam generator feedwater ring inspection	Yes, plant specific (See Subsection 3.1.2.2.14)
BWR/PWR	Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No
BWR/PWR	CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No
BWR/PWR	CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No
BWR/PWR	BWR piping and fittings; steam generator components	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/ PWR	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No
BWR	Feedwater and control rod drive (CRD) return line nozzles	Crack initiation and growth due to cyclic loading	Feedwater nozzle; CRD return line nozzle	No
BWR	Vessel shell attachment welds	Crack initiation and growth due to SCC, IGSCC	BWR vessel ID attachment welds; water chemistry	No
BWR	Nozzle safe ends, recirculation pump casing, connected systems piping and fittings, body and bonnet of valves	Crack initiation and growth due to SCC, IGSCC	BWR stress corrosion cracking; water chemistry	No
BWR	Penetrations	Crack initiation and growth due to SCC, IGSCC, cyclic loading	BWR penetrations; water chemistry	No
BWR	Core shroud and core plate, support structure, top guide, core spray lines and spargers, jet pump assemblies, control rod drive housing, nuclear instrumentation guide tubes	Crack initiation and growth due to SCC, IGSCC, IASCC	BWR vessel internals; water chemistry	No
BWR	Core shroud and core plate access hole cover (welded and mechanical covers)	Crack initiation and growth due to SCC, IGSCC, IASCC	ASME Section XI inservice inspection; water chemistry	No
BWR	Jet pump assembly castings; orificed fuel support	Loss of fracture toughness due to thermal aging and neutron embrittlement	Thermal aging and neutron irradiation embrittlement	No

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR	Unclad top head and nozzles	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	No
PWR	CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy nozzles and penetrations; water chemistry	No
PWR	Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except CASS and bolting)	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No
PWR	Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No
PWR	External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material due to boric acid corrosion	Boric acid corrosion	No
PWR	Steam generator secondary manways and handholds (CS)	Loss of material due to erosion	Inservice inspection	No
PWR	Reactor internals, reactor vessel closure studs, and core support pads	Loss of material due to wear	Inservice inspection	No
PWR	Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No
PWR	Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No
PWR	Reactor vessel internals in fuel zone region (except Westinghouse and Babcock & Wilcox [B&W] baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement, and void swelling	PWR vessel internals; water chemistry	No

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	Steam generator upper and lower heads; tubesheets; primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC. IASCC	Inservice inspection; water chemistry	No
PWR	Vessel internals (except Westinghouse and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No
PWR	Reactor internals (B&W screws and bolts)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No
PWR	Reactor vessel closure studs and stud assembly	Loss of material due to wear	Reactor head closure studs	No
PWR	Reactor internals (Westinghouse upper and lower internal assemblies; CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System

Program	Description of Program	Implementation Schedule*
ISI	The program consists of periodic volumetric, surface, and/or visual examination of components and their supports for assessment, signs of degradation, and corrective actions. This program is in accordance with ASME Section XI, 1995 edition through the 1996 addenda.	Existing program
Water chemistry	To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits based on EPRI guidelines of TR-103515 for water chemistry in BWRs, TR-105714 for primary water chemistry in PWRs, and TR-102134 for secondary water chemistry in PWRs.	Existing program
One-time inspection	To verify the effectiveness of the water chemistry control program by determining if the aging effect is not occurring or the aging effect is progressing slowly so that the that the intended function will be maintained during the period of extended operation, a one-time inspection of small-bore piping less than NPS 4, including pipe, fittings, and branch connections, using suitable techniques at the most susceptible locations is performed. Actual inspection locations should be based on physical accessibility, exposure levels, and NDE techniques, and locations identified in NRC IN 97-46.	Inspection should be completed before the period of extended operation.
Bolting integrity	This program consists of guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and enhanced inspection techniques. This program relies on the bolting integrity program delineated in NUREG-1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting and in EPRI TR-104213 for pressure retaining bolting and structural bolting.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
Reactor vessel surveillance	Periodic testing of metallurgical surveillance samples is used to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, Rev. 2.	The surveillance capsule withdrawal schedule should be revised before the period of extended operation.
Boric acid corrosion	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy. This program is implemented in response to GL 88-05.	Existing program
Thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel	The program consists of (1) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary.	Program should be implemented before the period of extended operation.
Reactor Head Closure Studs	This program includes inservice inspection ISI. For boiling water reactors (BWRs), this program also includes additional preventive actions and inspection techniques.	Existing program
Flow-accelerated corrosion	The program consists of the following: (1) conduct appropriate analysis and baseline inspection, (2) determine extent of thinning and replace/repair components, and (3) perform follow up inspections to confirm or quantify and take longer-term corrective actions. This program is in response to NRC GL 89-08.	Existing Program
Quality assurance	The 10 CFR Part 50, Appendix B program provides for corrective actions, confirmation process, and administrative controls for aging management programs for license renewal. The scope of this existing program will be expanded to include nonsafety-related structures and components that are subject to an AMR for license renewal.	Program should be implemented before the period of extended operation.
Vessel closure head penetration	The program assesses degradation of CRD mechanism nozzle and other vessel closure head penetrations, and consists of a review of the scope and schedule of inspection, including the leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. This is in response to NRC GL 97-01.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
BWR Control Rod Drive Return Line Nozzle	The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with the NUREG-0619 and alternative recommendation of GE NE-523-A71-0594. NUREG-0619 specifies UT of the entire nozzle and penetration testing (PT) of varying portions of the blend radius and bore. GE NE-523-A71-0594 specifies UT of specific regions of the blend radius and bore. UT techniques and personnel qualification are according to the guidelines of GE NE-523-A71-0594.	Program should be implemented before the period of extended operation.
Steam generator tube integrity	This program consists of SG inspection scope, frequency, acceptance criteria for the plugging and repair of flawed tubes in accordance with the plant technical specifications that includes commitments to NEI 97-06.	Existing program
Loose part monitoring	The program consists of loose part monitoring of reactor vessel and primary coolant systems in accordance with ASME OM-S/G-1997 standards. The program addresses methods, intervals, parameters to be measured and evaluated, and records requirements.	Existing program
Neutron noise monitoring	The program consists of neutron noise monitoring for the detection of loss of axial preload at the core support barrel's upper support flange, and can detect physical displacement and motion of reactor internals in accordance with ASME OM-S/G-1997 standards. The program addresses methods, intervals, parameters to be measured and evaluated, acceptance criteria, and records requirements.	Existing program
BWR Vessel Internals	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.	Existing program
Plant-specific AMP	The description should contain information associated with the basis for determining that aging effects will be managed during the period of extended operation.	Program should be implemented before the period of extended operation.

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
BWR Vessel ID Attachment Welds	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-48 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515).	
BWR Stress Corrosion Cracking	The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS) is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The program includes (a) preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects.	Existing program
BWR Penetrations	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-49 and BWRVIP-27 documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.	Existing program
Nickel-Alloy Nozzles and Penetrations	The program includes (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection ISI of reactor vessel head penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter (GL) 97-01.	Existing program
Thermal Aging of Cast Austenitic Stainless Steel	This program includes (a) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement and (b) for potentially susceptible components aging management is accomplished through either enhanced volumetric examination or plant- or component-specific flaw tolerance evaluation.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
PWR Vessel Internals	The program includes (a) augmentation of the inservice inspection (ISI) to include enhanced VT-1 examinations of non-bolted components, and other demonstrated acceptable methods for bolted components for certain susceptible or limiting components or locations, and (b) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of pressurized water reactor (PWR) vessel internal components.	Program should be implemented before the period of extended operation.
BWR Feedwater Nozzle	This program includes (a) enhancing inservice inspection (ISI) specified in the American Society of Mechanical Engineers (ASME) Code, Section XI, with the recommendation of General Electric (GE) NE-523-A71-0594 to perform periodic ultrasonic testing inspection of critical regions of the BWR feedwater nozzle.	Existing program

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

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