3.1 AGING MANAGEMENT OF 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 of the Reactor Coolant System. For a recent vintage plant, the information related to the 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). The Reactor Coolant System consists of the reactor vessel and internals and, for BWRs, 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 when generic existing programs are adequate to manage aging without change and when 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

An applicant may reference the GALL report in a license renewal application to demonstrate that the applicant's programs at its facility correspond to those reviewed and approved in the GALL 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 GALL 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 GALL report. Rather, the staff should ensure 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.

The BWR Vessel Improvement Program (BWRVIP) is currently under staff review. The August 2000, draft GALL report indicates that the BWRVIPs that have not been approved require further evaluation. If these BWRVIPs are approved, the applicant may reference the BWRVIP, as approved in the topical report, and no further staff evaluation is required. Because the reviews of these BWRVIP reports are scheduled to be completed before the final issuance of this standard review plan, the staff has not developed separate review guidance in this draft standard review plan.

3.1.1.3 Aging Management Programs or Evaluations that Are Different from those Described in the GALL Report

The GALL report provides a generic staff evaluation of certain aging management programs. If an 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 aging management program to which the GALL report does not apply.

3.1.1.4 Components or Aging Effects that Are Not Addressed in the GALL Report

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

3.1.1.5 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 define methods for determining if the applicant has met the requirements of the Commission’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 Coolant System are described and evaluated in Chapter IV of the GALL report (Ref. 2). In referencing the GALL report, an applicant should indicate that the material presented in the GALL report is applicable to the specific plant involved and provide the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report. An applicant should also verify that the approvals set forth in the GALL report for generic programs apply to the applicant’s programs. An 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:

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 of this standard review plan.

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

Loss of material due to pitting and crevice corrosion could occur in the steam generator shell assembly. 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, based on NRC Information notice 90-04 where 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.

Loss of material due to pitting and crevice corrosion and crack initiation and growth due to unanticipated thermal and mechanical loading or stress corrosion cracking could occur in BWR isolation condenser components. The existing AMP relies on control of reactor water chemistry to mitigate corrosion and ASME Section XI inservice inspection to detect leakage. However, the AMP is inadequate to detect cracking due to stress corrosion cracking and cyclic loading or loss of material due to pitting and crevice corrosion. The GALL report recommends a program consisting of 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.

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

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 of this standard review plan.

Neutron embrittlement of the reactor vessel is monitored by a reactor vessel materials surveillance program. The GALL report recommends further evaluation of this surveillance program for the period of extended operation.

Loss of fracture toughness due to neutron irradiation embrittlement could occur in PWR reactor vessel internals (except Westinghouse and B&W baffle/former bolts) and vessel beltline shell and welds. Loss of fracture toughness is a consequence only if cracks exist. Because cracking is expected to initiate at the surface, the ASME Section XI inservice inspection (ISI) relies on visual VT-3 examination to detect cracks. However, visual VT-3 examination is inadequate for creviced regions or for detecting tight cracks, and enhanced inspection and supplementary UT or other nondestructive examinations may be needed for effective means of detecting cracks and ensure that the component intended function will be maintained during the extended period.

The GALL report recommends enhanced inservice inspection to detect tight cracks and supplemental examinations for creviced regions.

Loss of fracture toughness due to neutron irradiation embrittlement 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 Unanticipated Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR/PWR)

Crack initiation and growth due to unanticipated thermal and mechanical loading or stress corrosion cracking could occur in small-bore reactor coolant system and connected system piping. The AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate SCC. However, Section XI inspection does not require volumetric examination of pipes less than NPS 4, and enhanced inspection and a one-time inspection of most susceptible locations is needed to ensure that degradation is not occurring and that the component's intended function will be maintained during the period of extended operation. The GALL report recommends enhanced inspection and a one-time inspection of a sample of locations is an acceptable method to ensure that this aging effect is not occurring and the component's intended function will be maintained during the period of extended operation.

Crack initiation and growth due to unanticipated thermal and mechanical loading or stress corrosion cracking (including IGSCC) could occur in BWR reactor vessel flange leak detection line, BWR jet pump sensing line and BWR separator support ring. The GALL report recommends further evaluation to ensure these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1, Appendix A.1 of this standard review plan.

3.1.2.2.5 Crack Initiation and Growth Due to Stress Corrosion Cracking or Intergranular Stress Corrosion Cracking (BWR)

Crack initiation and growth due to stress corrosion cracking or intergranular stress corrosion cracking could occur in BWR shroud support structure. The inspection and flaw evaluation guidelines of BWRVIP-38 for shroud support are as described in the staff approved topical report. This BWRVIP requires a plant-specific program. The GALL report recommends further evaluation of this plant-specific program.

3.1.2.2.6 Changes in Dimension Due to Void Swelling (BWR/PWR)

Changes in dimension due to void swelling could occur in reactor internal components. The GALL report recommends further evaluation to ensure 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. An acceptable AMP consists of participation in industry programs to address the significance of change in dimensions due to void swelling and implementation of an inspection program should the results of the industry programs indicate the need for such inspections. 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)

Crack initiation and growth due to stress corrosion cracking (SCC) and primary water stress corrosion cracking (PWSCC) could occur in PWR core support pads, reactor vessel penetrations, pressurizer spray heads, flange leak detection line and steam generator instrument and drain nozzles. The GALL report recommends further evaluation to ensure these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1, Appendix A.1 of this standard review plan.

Crack initiation and growth due to SCC could occur in PWR cast austenitic stainless steel (CASS) or coolant system piping. 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.

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 AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate PWSCC. However, the program is inadequate to manage the effects of SCC on the intended function of Ni-alloy components. The GALL report recommends that the applicant should evaluate the susceptibility of Ni-alloys to PWSCC, perform a susceptibility study of all Ni-alloy components to identify the most susceptible locations and to determine whether an augmented inspection program, including a combination of surface and volumetric examination, is necessary. Based on GL 97-01, the applicant should review the scope and schedule of inspection, including leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. The GALL report recommends that the applicant should either provide the technical basis that justifies the adequacy of the program or develop an integrated long-term program which includes periodic inspection of the most susceptible locations to detect the occurrence of PWSCC.

Crack initiation and growth due to SCC could occur in PWR primary nozzles and safe ends. The existing AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate SCC. The extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. Also, the applicant should review Ni-alloy applications in primary coolant and implement an augmented inspection program and evaluate choice of transducers for ultrasonic examination of dissimilar metal welds based on information provided in NRC Information Notices (INs) 90-10 and 90-30.

Crack initiation and growth due to SCC could occur in once-through steam generator upper and lower head, tube sheets and primary nozzles and safe ends. Aging management programs for controlling SCC in once-through steam generator upper and lower head, tube sheets and primary nozzles and safe ends include inservice inspections, guidelines to avoid sensitization of the stainless steel cladding, and primary water chemistry guidelines. However, the GALL report indicates that ASME Section XI requires inservice inspection of only the welds and weld regions and does not address the potential for cladding cracking remote from the welds. The GALL report recommends enhanced inspection and supplemental methods to detect cracking and ensure that the component intended function will be maintained during the period of extended operation.

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 stress corrosion cracking (SCC) or irradiation-assisted stress corrosion cracking (IASCC) could occur in PWR reactor vessel internals (all designs). The AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate SCC or IASCC. However, visual VT-3 examination may not be adequate for creviced regions or to detect tight cracks, and enhanced inspection techniques and supplemental techniques are needed to ensure that the component intended function will be maintained during the extended period. Therefore, the GALL report recommends enhanced inspection techniques and supplemental techniques to ensure that the component intended function will be maintained during the extended period.

Crack initiation and growth due to SCC or IASCC could occur in Westinghouse and B&W 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 have identified cracking in several plants. 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 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 PWR reactor vessel internal bolts and screws of B&W design. The ASME Section XI inspection relies on visual VT-3 examination to reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, or debris. However, visual VT-3 examination may not be adequate to detect loss of mechanical closure integrity, and enhanced inspection techniques and augmented inspection program are needed to ensure that the component intended function will be maintained during the extended period. The GALL report recommends enhanced inservice inspection to detect loss of mechanical closure integrity and augmented inspection program to determine critical locations and monitoring techniques.

Loss of preload due to stress relaxation could occur in baffle/former bolts in Westinghouse and B&W reactors. Because only heads of the baffle/former bolts are visible, the ASME Section XI visual VT-3 examination is inadequate to detect relevant conditions of stress relaxation. The GALL report recommends further evaluation to ensure 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 Wall Thinning Due to Erosion (PWR)

Wall thinning due to erosion could occur in steam generator feedwater impingement plate and support. The GALL report recommends further evaluation to ensure 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 Quality Assurance for Aging Management of Non-Safety-Related Components (BWR/PWR)

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

3.1.2.3 Aging Management Programs or Evaluations that Are Different from those Described 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 Components or Aging Effects that Are Not Addressed in the GALL Report

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

3.1.2.5 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 provide appropriate description 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

An 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 GALL 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 GALL 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 had 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 Coolant System components that are contained in the GALL 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 applicant may state that certain of its aging management programs contain the same program elements as the corresponding generic program described in the GALL report and upon which the staff relied in its evaluation, as described in the GALL report in accepting the generic program. The applicant may then state 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 GALL report regarding the Reactor Coolant System components are tabulated in Table 3.1-1 of this review plan section. 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 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 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)

The GALL report recommends further evaluation for the management of loss of material due to pitting and crevice corrosion of the steam generator shell assembly. Based on NRC Information notice 90-04 where 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.

The management of loss of material due to pitting and crevice corrosion and crack initiation and growth due to unanticipated thermal and mechanical loading, and stress corrosion cracking of BWR isolation condenser components should be further evaluated. The GALL report recommends an augmented inspection program. The existing AMP relies on control of reactor water chemistry to mitigate corrosion and ASME Section XI inservice inspection to detect leakage. However, the inspection requirements are inadequate to detect general corrosion and pitting, and 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 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.

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

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.

The GALL report recommends further evaluation of the reactor vessel materials surveillance program for the period of extended operation. 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.

Loss of fracture toughness due to neutron irradiation embrittlement could occur in PWR reactor vessel internals in the fuel zone region (all designs) except Westinghouse and B&W baffle/former bolts. The GALL report recommends enhanced inservice inspection to detect tight cracks and supplemental examinations for creviced regions. Loss of fracture toughness is a consequence only if cracks exist. Because cracking is expected to initiate at the surface, the ASME Section XI inspection relies on visual VT-3 examination to detect cracks. However, visual VT-3 examination is inadequate for creviced regions or for detecting tight cracks, and enhanced inspection and supplementary UT or other nondestructive examinations may be needed for effective means of detecting cracks and ensure that the component intended function will be maintained during the extended period. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage the loss of fracture toughness due to neutron irradiation embrittlement by providing enhanced inspection and supplemental methods to detect cracks and ensure that the component intended function will be maintained during the extended period.

The GALL report recommends further evaluation for the management of loss of fracture toughness due to neutron irradiation embrittlement 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.

The GALL report recommends enhanced inservice inspection to detect tight cracks and supplemental examinations for creviced regions of reactor pressure vessel beltline shell and nozzles. Loss of fracture toughness is a consequence only if cracks exist. VT-3 examination is inadequate for creviced regions or for detecting tight cracks. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that provides enhanced inspection and supplemental methods to detect cracks and ensure that the component intended function will be maintained during the extended period.

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

The management of crack initiation and growth due to unanticipated thermal and mechanical loading or stress corrosion cracking of small-bore reactor coolant system and connected system piping. The GALL report recommends enhanced inspection and a one-time inspection of a sample of locations. The AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate SCC. However, Section XI inspection does not require volumetric examination of pipes less than NPS 4, and enhanced inspection and a one-time inspection of select components and susceptible locations is an acceptable method to ensure that degradation is not occurring and that the component's intended function will be maintained during the period of extended operation. The staff reviews the applicant's proposed program to ensure that crack initiation and growth is not occurring and the component's intended function will be maintained during 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 would be performed using techniques similar to ASME Code and ASTM standards including visual, ultrasonic, and surface techniques (Ref. 6-7).

The GALL report recommends further evaluation of the management of crack initiation and growth due to unanticipated thermal and mechanical loading or stress corrosion cracking of reactor vessel flange leak detection line, and BWR jet pump sensing line and separator support ring. 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 Initiation and Growth Due to Stress Corrosion Cracking or Intergranular Stress Corrosion Cracking (BWR)

The management of crack initiation and growth due to stress corrosion cracking or intergranular stress corrosion cracking of BWR shroud support structure is to be further evaluated. The GALL report recommends the evaluation described in BWRVIP-38. The inspection and flaw evaluation guidelines of BWRVIP-38 for shroud support are as described in the staff approved topical report. 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.

3.1.3.2.6 Changes in Dimension Due to Void Swelling (BWR/PWR)

The GALL report recommends further evaluation of programs to manage changes in dimension due to void swelling for reactor internal components. 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. An acceptable AMP consists of participation in industry programs to address the significance of change in dimensions due to void swelling and implementation of an inspection program should the results of the industry programs indicate the need for such inspections. 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 provides 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)

The GALL report recommends further evaluation of programs to manage crack initiation and growth due to stress corrosion cracking (SCC) and primary water stress corrosion cracking (PWSCC) of PWR core support pads, reactor vessel penetrations, pressurizer spray heads, flange leak detection line and steam generator instrument and drain nozzles should be further evaluated. 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.

The GALL report recommends further evaluation of programs to manage crack initiation and growth due to SCC of PWR cast austenitic stainless steel (CASS) or coolant system piping. The GALL report recommends a plant specific aging management program for piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313. Crack initiation and growth from SCC could be significant for piping that does not meet the water chemistry or material guidelines and a plant specific aging management program must be evaluated. 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.

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 AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate PWSCC. However, the program is inadequate to manage the effects of SCC on the intended function of Ni-alloy components. GALL recommends that the applicant should evaluate the susceptibility of Ni-alloys to PWSCC. Based on the experiences described in NRC Information Notices (IN) 90-10 and 96-11, and Generic Letter (GL) 97-01, the susceptibility of Ni-alloys and welds to primary water PWSCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. The AMP monitors the effects of PWSCC on the intended function of instrument nozzles and penetrations by detection of cracks and leakage by ISI. Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Examination category B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. However, the applicant should perform a susceptibility study of all Ni-alloy components to identify the most susceptible locations and to determine whether an augmented inspection program, including a combination of surface and volumetric examination, is necessary. Aging effects degradation of the vessel penetrations cannot occur without crack initiation. Based on GL 97-01, the applicant should review the scope and schedule of inspection, including leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. The applicant should either provide the technical basis that justifies the adequacy of the program or develop an integrated long-term program which includes periodic inspection of the most susceptible locations to detect the occurrence of PWSCC. The frequency of subsequent inspections should be based on the finding of the initial inspections and crack growth rate models for Ni alloys. Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3500. However, applicant should provide information on crack initiation and growth models and the data used to validate these models to verify adequacy of the inspection program and acceptance criteria. 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.

The GALL report recommends further evaluation of programs to manage crack initiation and growth due to SCC of PWR primary nozzles and safe ends. The existing AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate SCC. However, the extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. Also, applicant should review Ni-alloy applications in primary coolant and implement an augmented inspection program and evaluate choice of transducers for ultrasonic examination of dissimilar metal welds based on information in NRC Information Notices (Ins) 90-10 and 90-30. The staff reviews the applicant’s program to verify that these recommendations for review and augmented inspection are effectively implemented to ensure that an adequate program will be in place for the management of these aging effects.

The GALL report recommends further evaluation for controlling SCC and IGSCC in once-through steam generator upper and lower head, tube sheets and primary nozzles and safe ends. Exist programs include inservice inspections, guidelines to avoid sensitization of the stainless steel cladding, and primary water chemistry guidelines. However, the GALL report indicates that ASME Section XI requires inservice inspection of only the welds and weld regions and does not address the potential for cladding cracking remote from the welds. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage crack initiation and growth by providing enhanced inspection and supplemental methods to detect cracking and ensure that the component intended function will be maintained during the extended period.

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

The management of crack initiation and growth due to stress corrosion cracking (SCC) or irradiation-assisted stress corrosion cracking (IASCC) of PWR reactor vessel internals (all designs) should be further evaluated. The GALL report recommends enhanced inspection techniques and supplemental techniques to ensure that the component intended function will be maintained during the extended period. The AMP relies on ASME Section XI inservice inspection to detect cracks and control of water chemistry to mitigate SCC or IASCC. However, visual VT-3 examination may not be adequate for creviced regions or to detect tight cracks, and enhanced inspection techniques and supplemental techniques are needed to ensure that the component intended function will be maintained during the extended period. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage crack initiation and growth due to SCC or IASCC by providing enhanced inspection and supplemental methods to detect cracks and ensure that the component intended function will be maintained during the extended period.

The GALL report recommends further evaluation of crack initiation and growth due to SCC or IASCC of Westinghouse and B&W 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 have identified cracking in several plants. 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 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 management of loss of preload due to stress relaxation of PWR reactor vessel internal bolts and screws of B&W design should be further evaluated. The GALL report recommends enhanced inservice inspection to detect loss of mechanical closure integrity and augmented inspection program to determine critical locations and monitoring techniques. The ASME Section XI inspection relies on visual VT-3 examination to reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, or debris. However, visual VT-3 examination may not be adequate to detect loss of mechanical closure integrity, and enhanced inspection techniques and augmented inspection program are needed to ensure that the component intended function will be maintained during the extended period. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage crack initiation and growth due to loss of preload from stress relaxation by providing enhanced inspection and supplemental methods to detect cracks and ensure that the component intended function will be maintained during the extended period.

The GALL report recommends further evaluation of loss of preload due to stress relaxation of baffle/former bolts in Westinghouse and B&W reactors should be further evaluated. Because only heads of the baffle/former bolts are visible, the ASME Section XI visual VT-3 examination is inadequate to detect relevant conditions of stress relaxation. 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 Wall Thinning Due to Erosion

The management of wall thinning due to erosion of steam generator feedwater impingement plate and support should be further evaluated. 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 Quality Assurance for Aging Management of Non-Safety-Related Components

An applicant's aging management programs for license renewal should contain the elements of corrective actions, 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 non-safety-related components that are subject to an aging management review for license renewal. Nevertheless, an 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 an applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If an applicant chooses other 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 Programs or Evaluations that Are Different from those Described 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 Components or Aging Effects that Are Not Addressed in the GALL Report

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

3.1.3.5 FSAR Supplement

The reviewer verifies that the applicant has provided information to be included in the FSAR supplement for aging management of the Reactor Coolant System for license renewal with information equivalent to that in Table 3.1-2 of this review plan section. The reviewer also verifies that the applicant has provided information to be included in the FSAR supplement for Subsection 3.1.3.3, "Aging Management Programs or Evaluations that are Different from those Described in the GALL Report," and Subsection 3.1.3.4, "Components or Aging Effects that are Not Addressed in the GALL Report," of this review plan section with information equivalent to that in Table 3.1-2. The staff expects to impose a license condition in the renewed license, if granted, 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 Commission 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, an applicant should identify and commit to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition in the renewed license, if granted, 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 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 Coolant System.

3.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission 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-xxxx, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, XXXX.

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.

Table 3.1-1. Aging Management Programs for 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 from pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects should be further evaluated (see subsection 3.1.2.2.2)
BWR	Isolation condenser	Crack initiation and growth from SCC and unanticipated cyclic loading, Loss of material from general, pitting, and crevice corrosion	Inservice inspection; water chemistry	Yes, plant specific (see subsection 3.1.2.2.2)
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 from neutron irradiation embrittlement	TLAA evaluated in accordance with Appendix G and H of 10 CFR50 and RG 1.99	Yes; TLAA (see subsection 3.1.2.2.3)
BWR/ PWR	Reactor vessel beltline shell and welds	Loss of fracture toughness from neutron irradiation embrittlement	Reactor vessel materials surveillance	Yes, plant specific (see subsection 3.1.2.2.3)
PWR	Reactor vessel internals in fuel zone region (except Westing-house and B&W baffle bolts)	Loss of fracture toughness from neutron irradiation embrittlement	Inservice inspection	Yes, detection of aging effects should be further evaluated (see subsection 3.1.2.2.3)
PWR	Westinghouse and B&W baffle/ former bolts	Loss of fracture toughness from neutron irradiation embrittlement	Plant-specific	Yes; plant-specific (see subsection 3.1.2.2.3)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/ PWR	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth from SCC and unanticipated cyclic loading	Inservice inspection; water chemistry; one-time inspection	Yes, detection of aging effects should be further evaluated. (see subsection 3.1.2.2.4)
BWR	Jet pump sensing line, Reactor Vessel flange leak detection line, and separator support ring	Crack initiation and growth from SCC, IGSCC or unanticipated cyclic loading	Plant specific	Yes, plant specific (see subsection 3.1.2.2.4)
BWR	Shroud Support Structure	Crack initiation and growth from SCC, IGSCC, and IASCC	Inservice Inspection; BWRVIP-38; water chemistry	Yes, plant specific BWRVIP-38 (see subsection 3.1.2.2.5)
PWR	Reactor internals	Changes in dimension from void swelling	Plant-specific	Yes; plant-specific (see subsection 3.1.2.2.6)
PWR	Reactor vessel flange leak detection line, core support pads, reactor vessel penetrations, pressurizer spray head, SG Instrument and drain nozzles,	Crack initiation and growth from SCC and/or PWSCC	Plant-specific	Yes; plant-specific (see subsection 3.1.2.2.7)
PWR	Cast austenitic SS reactor coolant system piping	Crack initiation and growth from SCC	Plant-specific	Yes; plant-specific (see subsection 3.1.2.2.7)
PWR	Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth from PWSCC	Inservice inspection; water chemistry	Yes, plant-specific (see subsection 3.1.2.2.7)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	Primary Nozzles and Safe Ends	Crack initiation and growth from SCC	Inservice inspection; water chemistry	Yes, detection of aging effects should be further evaluated (see subsection 3.1.2.2.7)
PWR	Upper & Lower Heads, Tube Sheets	Crack initiation and growth from SCC	Inservice inspection; water chemistry	Yes, detection of aging effects should be further evaluated (see subsection 3.1.2.2.7)
PWR	Vessel internals (except Westing-house and B & W baffle former bolts)	Crack initiation and growth from SCC and IASCC	Inservice inspection; water chemistry	Yes, detection of aging effects should be further evaluated (see subsection 3.1.2.2.8)
PWR	Westinghouse and B & W baffle former bolts	Crack initiation and growth from 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 from stress relaxation	Plant-specific	Yes; plant-specific (see subsection 3.1.2.2.9)
PWR	Reactor internals (B&W screws and bolts)	Loss of preload from stress relaxation	Inservice inspection	Yes, detection of aging effects should be further evaluated (see subsection 3.1.2.2.9)
PWR	Steam generator feedwater impingement plate and support	Loss of section thickness from erosion	Plant specific program	Yes, plant specific (see subsection 3.1.2.2.10)
BWR/ PWR	Reactor vessel closure studs and stud assembly	Crack initiation and growth from SCC and/or IGSCC	Inservice inspection; Minimization and control of SCC	No
BWR/ PWR	Cast austenitic stainless steel (CASS) pump casing and valve body	Loss of fracture toughness from thermal aging embrittlement	Inservice inspection	No

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/ PWR	Cast austenitic stainless steel (CASS) piping	Loss of fracture toughness from thermal aging embrittlement	Thermal aging and neutron embrittlement	No
BWR/ PWR	BWR piping and fittings, Steam generator components	Wall thinning from flow accelerated corrosion	Flow accelerated corrosion	No
BWR/PWR	RCPB valve closure bolting, man way and holding bolting, Closure bolting in high-pressure and high-temperature systems	Loss of material from atmospheric corrosion, loss of preload from stress relaxation, and crack initiation and growth from cyclic loading, stress corrosion cracking	Bolting integrity	No
BWR	Feedwater and CRD return line nozzles	Crack initiation and growth from cyclic loading	Feedwater and CRD inspection	No
BWR	Vessel Internals, Vessel Shell and Nozzle, and reactor coolant pressure piping	Crack initiation and growth from SCC, IGSCC, IASCC or unanticipated cyclic loading	Inservice Inspection; BWRVIP; water chemistry	No, (See statement in subsection 3.1.2.1.2)
BWR	Jet pump assembly castings, orificed fuel support	Loss of fracture toughness from thermal aging and neutron embrittlement	Thermal aging and neutron Irradiation embrittlement;	No
BWR	Unclad top head and nozzles	Loss of material from general, pitting, and crevice corrosion	Inservice inspection, water chemistry	No
PWR	Reactor vessel closure studs, and core support pads		Inservice inspection	No

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	CRD nozzle	Crack initiation and growth from PWSCC	Inservice inspection; vessel closure head penetration program; water chemistry	No
PWR	(Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth from PWSCC, ODSCC, and/or IGA or loss of material from general and pitting corrosion or deformation from corrosion at tube support plate intersections	Steam generator tube integrity; Water chemistry	No
PWR	Reactor vessel nozzles safe ends and CRD housing, reactor coolant system components (Except CASS and bolting)	Crack initiation and growth from unanticipated cyclic loading and/or SCC	Inservice inspection; water chemistry	No
PWR	Reactor vessel internals cast austenitic stainless steel components	Loss of fracture toughness from thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement; inservice inspection	No
PWR	External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material from boric acid corrosion	Boric acid corrosion	No
PWR	Steam generator secondary man ways and handholds (CS)	Loss of material from erosion	Inservice inspection	No

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR (CE)	Steam generator feedwater inlet ring	Loss of material from flow accelerated corrosion	CE steam generator feedwater ring inspection	No
PWR	Steam generator tubes	Loss of material from fretting and wear	Inservice inspection	No
PWR	Reactor internals, Reactor vessel closure studs, and core support pads	Loss of material from wear	Inservice inspection	No
PWR	Upper and lower internals assembly (Westinghouse)	Loss of preload from stress relaxation	Inservice inspection and loose part and/or neutron noise monitoring	No
PWR	Reactor internals (CE bolts, Tie Rods)	Loss of preload from stress relaxation	Inservice inspection and loose part monitoring	No

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Coolant System

Program	Description of Program	Implementation Schedule*
Inservice inspection	The program consists of periodic volumetric, surface, and/or visual examination of components and their supports for signs of degradation, assessment, and corrective actions. This program is in accordance with ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.	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. The water chemistry program relies on monitoring and control of water chemistry maintaining maximum 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 chemistry program, 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 to ensure that corrosion is not occurring. Actual inspection locations should be based on physical accessibility, exposure levels, and NDE examinations techniques, and locations identified in NRC Information Notice (IN) 97-46.	Inspection should be completed before the period of extended operation.
Minimization and control of SCC	This program consists of the guidelines of Regulatory Guide 1.65 on materials selection, materials properties, inspection, and protection against corrosion to minimize and control SCC problems in low-alloy steel reactor vessel closure bolting.	Existing program

Program	Description of Program	Implementation Schedule*
Fatigue monitoring program (FMP)	In order not to exceed the design limit on fatigue usage and the number of cycles, FMP monitors and tracks the number of critical thermal and pressure test transients, and monitors the cycles for the selected RCS components. The FPM will be modified to monitor a sample of components with high fatigue usage factors for the effects on the fatigue life. The FMP will assess the effect of the effect of the environment using statistical correlation developed by Argonne National Laboratory (ANL) in NUREG/CR-5704.	Program should be modified 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 EPRI NP-5067 for other bolting.	Existing program
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 1.99, Rev. 2. The withdrawal schedule will be revised to provide neutron fluence data at a neutron fluence equal to or greater than the projected peak fluence at the end of the license renewal period.	The surveillance capsule withdrawal schedule should be revised before the period of extended operation.

Program	Description of Program	Implementation Schedule*
Boric acid corrosion	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water for leaks, (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 and in accordance with ASME Section XI inservice inspection for reactor coolant leak tests.	Existing program
Thermal aging and neutron irradiation embrittlement	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.	Existing program
B & W reactor internal screw and bolt inspection	The program consists of inspection of B & W reactor internal screws and bolts (other than baffle former bolts) at critical locations using appropriate techniques to detect the loss of mechanical closure integrity.	Program should be implemented before the period of extended operation.
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 Generic Letter 89-08.	Program should be modified before the period of extended operation.
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 non-safety-related structures and components that are subject to an aging management review for license renewal.	Program should be implemented before the period of extended operation.

Program	Description of Program	Implementation Schedule*
Vessel closure head penetration	The program assesses degradation of control rod drive mechanism nozzle and other vessel closure head penetrations and consists of a review of the scope and schedule of inspection, including leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. This is in response to GL 97-01.	Existing program
Feedwater and CRD inspection	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 ultrasonic testing (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 examination techniques and personnel qualification is according to the guidelines of GE NE-523-A71-0594.	Program should be implemented before the period of extended operation.
Steam Generator Tube integrity	Aging management program for controlling PWSCC and ODSCC in steam generator tubes, sleeves, and plugs include inservice inspections in accordance with Plant Technical Specifications, EPRI document "PWR Steam Generator Examination Guidelines, Rev. 5" (Ref. 6), NEI 97-06 (Ref. 3), and NRC Regulatory Guide (RG) 1.83 (Ref. 7). Tube repairs should be in accordance with NRC RG 1.121 (Ref. 8), Generic Letter 95-05 (Ref. 9), or other NRC approved bases and the EPRI inspection guidelines and NEI 97-06 (Ref. 3).	Existing program

Program	Description of Program	Implementation Schedule*
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
BWRVIP	The BWR vessel internal program (VIP) consists of inspection, evaluation, maintenance, repair and water chemistry. The program addresses BWR reactor vessel, internals, and pressure boundary components. The BWRVIP is described in NRC approved topical reports.	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.
CE steam generator feedwater inlet ring inspection	The program monitors wall thinning and degradation of supports and attachments in accordance with the recommendations of CE info-bulletin 90.04.	Program should be implemented before the period of extended operation.

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