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

Item	Locator	Comment	Justification
1	3.1.2.2.2.1	Clarify intent of NRC Information Notice (IN) 90-04	Refer to comments for NUREG-1801 Vol. 2 Items IV.D1-16 and IV.D2-10. Discussion does not clearly convey the nature of the degradation mechanism, which is related to a very specific set of conditions, not to most SGs. IN 90-04 indicates that pitting corrosion on the surface served as corrosion fatigue crack initiation sites, not that pitting corrosion resulted in sufficient degradation to cause loss of component function. Further, this degradation mode has been limited to isolated cases of weld-zone cracking in Westinghouse Model 44 and 51 SGs, where a high stress region exists in the area of the shell to transition cone weld.
2	3.1.2.2.4.2	Delete redundant text	Editorial
3	3.1.2.2.4.3	Delete mention of cyclic loading and loss of material	The subject of this item is cracking due to SCC and IGSCC. As such, requirements to augment AMP for cracking due to cyclic loading or for loss of material due to pitting and crevice corrosion do not belong or are addressed elsewhere. Firstly, cyclic loading of isolation condenser components is not an aging effect identified in NUREG-1801 Volume 2. Secondly, NUREG-1801 Vol. 2 Item IV.C1-6 (R-16) addresses LOM. Table 3.1.1, Item 5 (R-16) lists 3.1.2.2.2.2 as the applicable SRP-LR section and not 3.1.2.2.4.3.
4	3.1.2.2.8	Delete “due to line” to correct sentence.	Editorial correction. Either delete words or determine what is missing from the 2nd sentence.
5	3.1.2.2.18	Eliminate specific fluence level	Editorial – fluence level is not required part of the description, neutron flux would be adequate terminology.
6	Table 3.1-1	Odd page footers are for NUREG-1801.	Editorial

7	Table 3.1-1, Table 3.2-1, Table 3.3-1, Table 3.4-1 Note: This comment actually applies to the rollup Tables of GALL Volume 1., which then become the tables in the SRP	When addressing Further Evaluation Recommended issues, avoid mixing internal and external environments for the same component. For example, Table 3.2-1, ID 2 addresses “Steel components (including piping and ducting) exposed to external condensation or outside air, internally or externally to indoor uncontrolled air.” Table 3.2-1, ID 11 addresses “Steel piping, piping components, and piping elements (internal surfaces) and ducting closure bolting exposed to condensation (internal, treated water, or air –indoor uncontrolled (external).” In this case, the words “internally or” should be deleted from ID 1 and the word “external” should be replaced with “internal.” Of course the related items from NUREG-1801, Volume 2, that deal with external surfaces should point to ID 1; while those dealing with internal surfaces should point to ID 11. If the NUREG-1801, Volume 2, related items are unclear about which surface is being addressed, those items should be clarified.	Typically the plant-specific program applied to internal surfaces is not the same program as applied to external surfaces. Addressing the plant-specific programs to be used would be simplified if separate LRA discussions were permitted for external and internal surfaces. The suggested changes would also eliminate a source of confusion regarding which surfaces are being addressed in GALL roll-up line items.
7	Table 3.1-1	Delete Loose Parts Monitoring and Neutron Noise rows from table.	These programs are not credited in GALL.

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

Review Responsibilities

Primary - Branch assigned responsibility by PM as described in SRP-LR section 3.0

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 an older plant’s FSAR may have predated NUREG-0800.

The reactor vessel, internals, and reactor coolant system includes the reactor vessel and internals. For BWRs, this system also includes 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 responsible review organization is to review the following LRA AMR and AMP items assigned to it, per SRP-LR section 3.0, for review:

<u>AMRs</u>

- AMRs consistent with the GALL report, for which further evaluation is not recommended
- AMRs consistent with the GALL report, for which further evaluation is recommended
- AMRs not consistent with the GALL report

<u>AMPs</u>

- AMPs consistent with GALL AMPs (with or without exceptions)
- Plant-specific AMPs

<u>FSAR Supplement</u>

- In addition, the responsible review organization is to review the FSAR supplement associated with each assigned AMP.

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 AMR Results Consistent with the GALL Report for Which No Further Evaluation is Recommended

The aging management review and acceptable aging management programs applicable to the reactor vessel, internals, and reactor coolant system are described and evaluated in Chapter IV of the GALL report (Ref. 2).

The applicant's LRA should provide sufficient information so that the NRC reviewer is able to confirm that the specific AMR line-item and the associated AMP are consistent with the cited GALL AMR line-item. The staff reviewer should then confirm that the LRA AMR line-item is consistent with the GALL line-item to which it is compared.

If the applicant identifies an exception to the cited GALL AMP, the LRA should include a basis demonstrating how the criteria of 10 CFR 54.21(a)(3) would still be met. The NRC reviewer should then confirm that the AMP with all exceptions would satisfy the criteria of 10 CFR 54.21(a)(3). If, while reviewing the AMP, the reviewer identifies a difference from the GALL AMP, this difference should be reviewed and dispositioned as if it were an exception identified by the applicant in its LRA. The disposition of all LRA-defined exceptions and staff-identified differences should be documented.

The LRA should identify any enhancements that are needed to permit an existing aging management program to be declared consistent with the GALL AMP to which the LRA AMP is compared. The reviewer is to confirm both that the enhancement, if implemented, would allow the existing plant aging management program to be consistent with the GALL AMP and also that the applicant has a commitment in the FSAR supplement to implement the enhancement prior to the period of extended operation. The reviewer should document the disposition of all enhancements.

3.1.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic acceptance criteria defined in 3.1.2.1 apply to all of the AMRs and AMPs reviewed as part of this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

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 General, Pitting, and Crevice Corrosion (BWR/PWR)

1. Loss of material due to general, pitting, and crevice corrosion could occur in the steel 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), the program may not be sufficient to detect pitting and crevice

corrosion***, if general and pitting corrosion of the shell is known to exist***. The GALL report recommends augmented inspection to manage this aging effect. ***Furthermore, the GALL Report clarifies that this issue is limited to Westinghouse Model 44 and 51 Steam Generators where a high stress region exists at the shell to transition cone weld.*** 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 stainless steel BWR isolation condenser components. General, pitting, and crevice corrosion could occur in steel 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 general, 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. Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all ferritic materials that have a neutron fluence greater than 10^{17} n/cm2 (E >1 MeV) at the end of the license renewal term. 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 BWR and PWR reactor vessels. 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 could occur in Westinghouse and B&W baffle/former bolts and screws. The GALL report recommends no further aging management review if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation~~,~~ submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.4 Cracking due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking (BWR)

1. Cracking due to stress corrosion cracking (SCC) and intergranular stress corrosion cracking [IGSCC]) could occur in small-bore steel and stainless steel reactor coolant system and connected system piping less than Nominal Pipe Size (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 is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. 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..

2. Cracking due to SCC and IGSCC could occur in the stainless steel and nickel alloy BWR reactor vessel flange leak detection lines. ~~The GALL report recommends~~ The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting cracking due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan)

3. Cracking due to SCC and IGSCC could occur in steel and stainless steel 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 ~~and cyclic loading or loss of material due to pitting and 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).

3.1.2.2.5 Crack Growth due to Cyclic Loading (PWR)

Crack growth due to cyclic loading could occur in reactor vessel shell forgings clad with stainless steel using a high-heat-input welding process. Growth of intergranular separations (underclad cracks) in the 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 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling (PWR)

Loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur in stainless steel, cast austenitic stainless steel, nickel alloy, and PH stainless steel forging reactor vessel internals. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.).

3.1.2.2.7 Cracking due to Stress Corrosion Cracking (PWR)

1. Cracking due to SCC could occur in the PWR stainless steel reactor vessel flange leak detection lines and bottom-mounted instrument guide tubes. 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 cracking due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Cracking due to SCC could occur in Class 1 PWR cast austenitic stainless steel (CASS) reactor coolant system piping, piping components, and piping elements. The GALL report recommends maintenance of reactor water chemistry in accordance with the guidelines of TR-105714 and further evaluation for piping, piping components, and piping elements that do not meet the material guidelines of NUREG-0313 (Ref. 5). For piping, piping components, and piping elements that do not meet the NUREG-0313 guidelines, the GALL report recommends that the program include (a) adequate inspection methods to ensure detection of cracks, and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Cracking due to stress corrosion cracking (SCC) could occur in PWR small-bore stainless steel 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 period of extended operation. The AMPs should be augmented by verifying that cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections.

3.1.2.2.8 Cracking due to Cyclic Loading (BWR)

Cracking due to cyclic loading could occur in the stainless steel BWR jet pump sensing lines. The GALL report recommends that a plant specific aging management program be evaluated to mitigate or detect cracking ~~due to line~~. 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 stainless steel and nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.10 Loss of Material due to Erosion (PWR)

Loss of material due to erosion could occur in steel 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 Cracking due to Flow-Induced Vibration (BWR)

Cracking due to flow-induced vibration could occur for the BWR stainless steel steam dryers. 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).

3.1.2.2.12 Cracking due to Thermal and Mechanical Loading (BWR/PWR)

Cracking due to thermal and mechanical loading could occur in Class 1 small-bore steel (BWR), steel with stainless steel cladding, and stainless steel reactor coolant system and connected system piping less than NPS 4. The existing program relies on ASME Section XI ISI to manage cracking due to thermal and mechanical loading. However, Inservice Inspection for Class 1 components Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. Therefore, a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. 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. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.13 Cracking due to Primary Water Stress Corrosion Cracking (PWR)

1. Cracking due to PWSCC could occur in PWR components made of nickel alloy or having nickel alloy cladding, including reactor coolant pressure boundary components and penetrations inside the RCS such as pressurizer heater sheathes and sleeves, nozzles, and other internal components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

2. Cracking due to PWSCC could occur in nickel alloy of PWR core support pads (or core guide lugs) and pressurizer spray heads. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to submit a plant-specific AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces.

3.1.2.2.14 Wall Thinning due to Flow-accelerated Corrosion (PWR)

1. Wall thinning due to flow-accelerated corrosion could occur in steel feedwater inlet rings 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 wall thinning due to flow-accelerated corrosion. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Wall thinning due to flow-accelerated corrosion could occur in the steel tube support lattice bars of PWR steam generators. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to submit, for NRC review and approval, an inspection plan for tube support lattice bars as based upon staff approved NEI 97-06 guidelines, or other alternative regulatory basis for steam generator degradation management, at least 24 months prior to the extended period.

3.1.2.2.15 Changes in Dimensions due to Void Swelling (PWR)

Changes in dimensions due to void swelling could occur in PWR reactor internal components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.16 Cracking due to Stress Corrosion Cracking and Primary Water Stress Corrosion Cracking (PWR)

Cracking due to SCC and PWSCC could occur in PWR stainless steel, cast austenitic stainless steel, nickel alloy, and steel with stainless steel or nickel alloy cladding pressurizer components; steam generator upper and lower heads, tubesheets, and CRD pressure housing components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

3.1.2.2.17 Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking (PWR)

Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking could occur in PWR stainless steel and nickel alloy reactor vessel internals. The existing program relies on control of water chemistry to mitigate these effects. However, the existing program should be augmented to manage these aging effects on the intended function of reactor vessel internals components. The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.18 Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (PWR)

1. Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking could occur in PWR stainless steel baffle former bolts and screws exposed to ***neutron flux*** ~~high fluence (>1 x 10e21 n/cm2, E >0.1 MeV)~~. The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

2. Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking could occur in PWR stainless steel, cast austenitic stainless steel, and PH stainless steel forging reactor vessel internals components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval

3.1.2.2.19 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 AMR Results Not Consistent with or Not Addressed in 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. The description should also contain any future aging management activities, including enhancements and commitments, to be completed before the period of extended operation. Examples of the type of information to be included are provided in Table 3.1-2 of this standard review plan.

3.1.3 Review Procedures

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

3.1.3.1 AMR Results Consistent with the GALL Report for which no Further Evaluation is Recommended

The applicant may reference the GALL report in its license renewal application, as appropriate, and demonstrate that the aging management reviews and programs at its facility are consistent with those reviewed and approved in the GALL report. The reviewer should not conduct a re-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 acceptable the applicant's reference to GALL in its license renewal application. 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 confirms 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.

Furthermore, the reviewer should confirm that the applicant has addressed operating experience identified after the issuance of the GALL report. Performance of this review requires the reviewer to confirm that the applicant has identified those aging effects for the reactor vessel, internals, and reactor coolant system components that are contained in GALL as applicable to its plant.

The reviewer confirms that the applicant has identified the appropriate AMPs as described and evaluated in the GALL report. If the applicant commits to an enhancement to make its aging management program consistent with a GALL AMP, then the reviewer is to confirm that this enhancement when implemented will indeed make the LRA AMP consistent with the GALL AMP. If an aging management program in the LRA identifies an exception to the GALL AMP to which it is claiming to be consistent, the reviewer is to confirm that the LRA AMP with the exception will satisfy the criteria of 10 CFR 54.21(a)(3). If the reviewer identifies a difference, not identified by the LRA, between the LRA AMP and the GALL AMP, with which the LRA claims to be consistent, the reviewer should confirm that the LRA AMP with this difference satisfies 10 CFR 54.21(a)(3). The reviewer should document the basis for accepting enhancements, exceptions or differences. The AMPs evaluated in GALL pertinent to the reactor vessel, internals, and reactor coolant system components are summarized in Table 3.1-1 of this standard review plan. In this table, the ID column provides a row identifier useful in matching the information presented in the corresponding table in the GALL report Vol. 1. The Related Item column identifies the item number in the GALL report Vol. 2, Chapters II through VIII, presenting detailed information summarized by this row.

3.1.3.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic review procedures defined in 3.1.3.1 apply to all of the AMRs and AMPs provided in this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

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 General, Pitting, and Crevice Corrosion (BWR/PWR)

1. The GALL report recommends further evaluation for the management of loss of material due to general, pitting, and crevice corrosion of the steel 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, pitting, and crevice corrosion of the shell exists, the existing program requirements may not be sufficient to detect loss of material due to these effects, and additional inspection procedures may be necessary. 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 period of extended operation.

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 general, 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 general, 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 general, pitting, and crevice corrosion and ensure that the component intended function will be maintained during the period of extended operation.

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 are provided in chapter XI, Section M31 of the GALL report.

3. The GALL report recommends no further evaluation for the management of loss of fracture toughness due to neutron irradiation embrittlement of Westinghouse and B&W baffle/former bolts and screws if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, 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 this aging effect.

3.1.3.2.4 Cracking due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking (BWR)

1. For steel and stainless steel BWR small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, the GALL report recommends Inservice Inspection for Class 1 components, Water Chemistry for BWR, and a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of component inside surfaces for the management of cracking due to SCC and IGSCC. Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4 and the existing program should be augmented by verifying that cracking is not occurring. 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 that a plant-specific aging management program is to be evaluated to manage cracking due to SCC and IGSCC in stainless steel and nickel alloy BWR reactor vessel flange leak detection lines. 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 cracking due to SCC, IGSCC and cyclic loading or loss of material due to pitting and crevice corrosion of the steel and stainless steel 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, IGSCC and cyclic loading or loss of material due to pitting and crevice corrosion. 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. Growth of intergranular separations (underclad cracks) in the 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 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling (PWR)

The GALL report recommends no further evaluation of programs to manage loss of fracture toughness due to neutron irradiation embrittlement and void swelling in stainless steel, cast austenitic stainless steel, nickel alloy, and PH stainless steel forging reactor vessel internals if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, 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.

3.1.3.2.7 Cracking due to Stress Corrosion Cracking (PWR)

1. The GALL report recommends that a plant-specific aging management program is to be evaluated to manage cracking due to stress corrosion cracking in stainless steel PWR reactor vessel flange leak detection lines and bottom-mounted instrument guide tubes. 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 maintenance of reactor water chemistry in accordance with the guidelines of TR-105714 and further evaluation for CASS piping, piping components, and piping elements that do not meet the material guidelines of NUREG-0313 (Ref. 5). For components that do not meet the NUREG-0313 guidelines, the GALL report recommends that the program include (a) adequate inspection methods to ensure detection of cracks, and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement. 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. For stainless steel PWR small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, the GALL report recommends Inservice Inspection for Class 1 components, maintenance of reactor water chemistry in accordance with the guidelines of TR-102134, Water Chemistry for PWR, and a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of component inside surfaces for the management of cracking due to SCC and IGSCC. Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4 and the existing program should be augmented by verifying that cracking is not occurring. 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 Cracking due to Cyclic Loading (BWR)

The GALL report recommends that a plant specific aging management program be evaluated for the management of cracking due to cyclic loading in stainless steel BWR jet pump sensing lines. 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 that could occur in stainless steel and nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, 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 this aging effect.

3.1.3.2.10 Loss of Material due to Erosion (PWR)

The GALL report recommends further evaluation of a plant-specific aging management program for the management of loss of material 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 Cracking due to Flow-Induced Vibration (BWR)

The GALL report recommends further evaluation of a plant-specific aging management program for the management of cracking due to flow-induced vibration of BWR stainless steel steam dryers. 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 Cracking due to Thermal and Mechanical Loading (BWR/PWR)

The GALL report recommends ASME Section XI ISI and a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of small-bore (less than NPS 4) Class 1 steel (BWR), steel with stainless steel clad, and stainless steel piping for the management of cracking due to thermal and mechanical loading. Inspection in accordance with ASME Section XI does not require volumetric examination of small-bore reactor coolant system and connected system piping less than NPS 4 and the existing program should be augmented by verifying that service-induced weld cracking is not occurring in the piping, fittings, and branch connections. 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. 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, should be performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50 Appendix B. Follow-up of unacceptable inspection findings should include expansion of the inspection sample size and locations. The staff reviews the program to confirm that it includes measures to determine if unacceptable degradation is occurring. If an applicant proposes a one-time inspection of select components and susceptible locations to ensure that cracking 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.

3.1.3.2.13 Cracking due to Primary Water Stress Corrosion Cracking (PWR)

1. The GALL report recommends no further aging management review is necessary to manage cracking due to PWSCC of PWR components made of nickel alloy or having nickel alloy cladding, including reactor coolant pressure boundary components and penetrations inside the RCS such as pressurizer heater sheathes and sleeves, nozzles, and other internal components, if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of PWSCC. If the FSAR supplement does not contain the appropriate commitment, 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 PWSCC.

2. The GALL report recommends no further aging management review is necessary to manage cracking due to PWSCC of PWR core support pads (or core guide lugs) and pressurizer spray heads if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of PWSCC. If the FSAR supplement does not contain the appropriate commitment, 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 PWSCC.

3.1.3.2.14 Wall Thinning due to Flow-accelerated Corrosion (PWR)

1. The GALL report recommends that a plant-specific aging management program be evaluated to manage loss of material due to wall thinning 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.

2. The GALL report recommends no further aging management review is necessary to manage wall thinning due to flow-accelerated corrosion in the steel tube support lattice bars of PWR steam generators if the applicant provides a commitment in the FSAR supplement to submit, for NRC review and approval, an inspection plan for tube support lattice bars as based upon staff approved NEI 97-06 guidelines, or other alternative regulatory basis for steam generator degradation management, at least 24 months prior to the extended period.

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of this aging effect. If the FSAR supplement does not contain the appropriate commitment, 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 this aging effect.

3.1.3.2.15 Changes in Dimensions due to Void Swelling (PWR)

The GALL report recommends no further aging management review is necessary to manage changes in dimensions due to void swelling in reactor internal components if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of this aging effect. If the FSAR supplement does not contain the appropriate commitment, 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 this aging effect.

3.1.3.2.16 Cracking due to Stress Corrosion Cracking and Primary Water Stress Corrosion Cracking (PWR)

The GALL report recommends no further aging management review is necessary to manage cracking due to SCC and PWSCC of PWR stainless steel, cast austenitic stainless steel, nickel alloy, and steel with stainless steel or nickel alloy cladding pressurizer components; steam generator upper and lower heads, tubesheets, and CRD pressure housing components if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of PWSCC. If the FSAR supplement does not contain the appropriate commitment, 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 SCC and PWSCC.

3.1.3.2.17 Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking (PWR)

The GALL report recommends further evaluation of programs to manage cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking in stainless steel and nickel alloy reactor vessel internals. The existing program relies on control of water chemistry to mitigate these effects. However, the existing program should be augmented to manage these aging effects on the intended function of reactor vessel internals components. The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, 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.

3.1.3.2.18 Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (PWR)

1. The GALL report recommends further evaluation of programs to manage cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking in stainless steel baffle former bolts and screws exposed to high fluence (>1 x 10e21 n/cm2, E >0.1 MeV). The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, 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 no further aging management review is necessary to manage cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking in stainless steel, cast austenitic stainless steel, and PH stainless steel forging reactor vessel internals components if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, 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.

3.1.3.2.19 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 AMR Results Not Consistent with or Not Addressed in GALL Report

The reviewer should confirm that the applicant, in the license renewal application, has identified applicable aging effects, listed the appropriate combination of materials and environments, and aging management programs that will adequately manage the aging effects. The aging management program credited could be an AMP that is described and evaluated in the GALL report or a plant-specific program. 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 confirms 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 confirms 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 confirm that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments 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

If the reviewer has confirmed that the applicant has provided information sufficient to satisfy the provisions of this review plan section, an evaluation finding similar to the following text should be included in the staff's safety evaluation report:

> On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects and the AMPs credited with managing these aging effects for the reactor vessel, internals and reactor coolant system, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB during the period of extended operation. The staff also reviewed the applicable FSAR supplement program descriptions and concludes that the FSAR supplement provides an adequate program description of the AMPs credited for managing aging effects, as required by 10 CFR 54.21(d).

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. Draft NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, Revision 1, September 2005.

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, April 1999.

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,* May 1988.

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.”

27. EPRI TR-102134, *PWR Secondary Water Chemistry Guideline-Revision 3*, Electric Power Research Institute, Palo Alto, CA, May 1993.

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
1	BWR/ PWR	Reactor coolant pressure boundary closure bolting, head closure studs, support skirts and attachment welds, pressurizer relief tank components, steam generator components, and reactor vessel internals	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (See subsection 3.1.2.2.1)	R-13 R-18 R-33 R-46 R-53 R-54 R-70 R-73 R-91
2	BWR/ PWR	Reactor coolant pressure boundary components, steam generator tubes and sleeves, reactor vessel internals, pressurizer components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c) and environmental effects are to be addressed for Class 1 components	Yes, TLAA (See subsection 3.1.2.2.1)	R-04 R-189 R-45
3	BWR/ PWR	Pump and valve closure bolting	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c) check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range	Yes, TLAA (See subsection 3.1.2.2.1)	R-28
4	PWR	Steel steam generator shell assembly	Loss of material due to general, pitting and crevice corrosion	Inservice inspection and water chemistry	Yes, detection of aging effects is to be evaluated (See subsection 3.1.2.2.2.1)	R-34
5	BWR	Stainless steel; steel isolation condenser tube side components exposed to reactor coolant	Loss of material due to general (steel only), pitting and crevice corrosion	Inservice inspection, water chemistry, and plant-specific verification program	Yes, detection of aging effects is to be evaluated (See subsection 3.1.2.2.2.2)	R-16

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
6	BWR/ PWR	Reactor vessel beltline shell, nozzles, and welds	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations.	Yes, TLAA (See subsection 3.1.2.2.3.1)	R-62 R-67 R-81 R-84
7	BWR/ PWR	Reactor vessel beltline shell, nozzles, and welds; safety injection nozzles	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific (See subsection 3.1.2.2.3.2)	R-63 R-82 R-86
8	PWR	Westinghouse stainless steel baffle former bolts	Loss of fracture toughness due to neutron irradiation embrittlement	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.3.3)	R-128
9	BWR	Steel and stainless steel Class 1 piping, fittings and branch connections < NPS 4 exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	Inservice Inspection, Water chemistry, and a plant specific examination	Yes, parameters monitored/inspected and detection of aging effects are to be evaluated (See subsection 3.1.2.2.4.1)	R-03

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
10	BWR	Stainless steel and nickel alloy reactor vessel flange leak detection line	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC in the vessel flange leak detection line.	Yes, plant specific (See subsection 3.1.2.2.4.2)	R-61
11	BWR	Stainless steel; steel isolation condenser tube side components exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	Inservice inspection, water chemistry, and plant-specific verification program	Yes, detection of aging effects is to be evaluated (See subsection 3.1.2.2.4.3)	R-15
12	PWR	Reactor vessel shell fabricated of SA508-Cl 2 forgings clad with stainless steel using a high-heat-input welding process	Crack growth due to cyclic loading	TLAA	Yes, TLAA (See subsection 3.1.2.2.5)	R-85

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
13	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy, and PH stainless steel forging reactor vessel internals components (except Westinghouse baffle former bolts) exposed to reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement, void swelling	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.6)	R-122 R-127 R-132 R-135 R-141 R-157 R-161 R-164 R-169 R-178 R-188 R-196 R-200 R-205 R-212 R-216
14	PWR	Stainless steel reactor vessel flange leak detection line and bottom-mounted instrument guide tubes	Cracking due to stress corrosion cracking	Plant specific	Yes, plant specific (See subsection 3.1.2.2.7.1)	R-74 RP-13
15	PWR	Class 1 cast austenitic stainless steel (CASS) piping, piping components, and piping elements	Cracking due to stress corrosion cracking	Water chemistry (and plant specific for components that do not meet the material guidelines of NUREG-0313).	Yes, plant specific for components that do not meet the material guidelines of NUREG-0313 (See subsection 3.1.2.2.7.2)	R-05

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
16	PWR	Stainless steel Class 1 piping, fittings and branch connections < NPS 4 exposed to reactor coolant	Cracking due to stress corrosion cracking	Inservice Inspection, Water chemistry, and a plant specific examination	Yes, parameters monitored/inspected and detection of aging effects are to be evaluated (See subsection 3.1.2.2.7.3)	R-02
17	BWR	Stainless steel jet pump sensing line	Cracking due to cyclic loading	Plant specific	Yes, plant specific (See subsection 3.1.2.2.8)	R-102
18	PWR	Stainless steel, nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs	Loss of preload due to stress relaxation	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.9)	R-108 R-114 R-129 R-136 R-137 R-154 R-165 R-184 R-192 R-197 R-201 R-207 R-213
19	PWR	Steel steam generator feedwater impingement plate and support	Loss of material due to erosion	Plant specific	Yes, plant specific (See subsection 3.1.2.2.10)	R-39
20	BWR	Stainless steel steam dryers exposed to reactor coolant	Cracking due to flow-induced vibration	Plant specific	Yes, plant specific (See subsection 3.1.2.2.11)	RP-18

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
21	BWR/ PWR	BWR steel and stainless steel Class 1 piping, fittings and branch connections < NPS 4 exposed to reactor coolant; PWR stainless steel and steel with stainless steel cladding Class 1 piping, fittings and branch connections < NPS 4	Cracking due to thermal and mechanical loading	Inservice Inspection and a plant specific examination (one-time inspection)	Yes, parameters monitored/inspected and detection of aging effects are to be evaluated (See subsection 3.1.2.2.12)	R-55 R-57
22	PWR	Nickel alloy reactor coolant pressure boundary penetrations; primary side nozzles and safe ends, pressurizer steam space nozzles, heater sheaths and sleeves, heater bundle diaphragm plate, and manways and flanges; steam generator divider plate	Cracking due to primary water stress corrosion cracking	Inservice Inspection and Water Chemistry, and for Alloy 600, FSAR supplement commitment to implement applicable plant commitments to (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed (See subsection 3.1.2.2.13.1)	R-01 R-06 R-218 R-75 R-89 R-90 RP-21 RP-22
23	PWR	Nickel alloy, cast austenitic stainless steel, stainless steel pressurizer spray head, core support pads/core guide lugs	Cracking due to primary water stress corrosion cracking	Water Chemistry and One-Time Inspection or Inservice Inspection and provide commitment in FSAR supplement to submit AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces.	No, unless licensee commitments need to be confirmed (See subsection 3.1.2.2.13.2)	R-24 R-88

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
24	PWR (CE)	Steel steam generator feedwater inlet ring and supports	Wall thinning due to flow-accelerated corrosion	Combustion Engineering (CE) System 80 steam generator feedwater ring inspection	Yes, plant specific (See subsection 3.1.2.2.14.1)	R-51
25	PWR	Steel tube support lattice bars exposed to secondary feedwater/ steam	Wall thinning due to flow-accelerated corrosion	FSAR supplement commitment to submit an inspection plan for tube support lattice bars for NRC review and approval, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.14.2)	R-41

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
26	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy, PH stainless steel forging reactor vessel internals components (except Westinghouse baffle former bolts)	Changes in dimensions due to void swelling	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.15)	R-107 R-110 R-113 R-117 R-119 R-121 R-124 R-126 R-131 R-134 R-139 R-144 R-147 R-151 R-158 R-160 R-163 R-168 R-174 R-177 R-182 R-187 R-195 R-199 R-204 R-211 R-215

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
27	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy, steel with stainless steel or nickel alloy cladding pressurizer components; steam generator upper and lower heads, tubesheets; CRD pressure housing	Cracking due to stress corrosion cracking, primary water stress corrosion cracking	Inservice Inspection and Water Chemistry, and for Alloy 600, FSAR supplement commitment to implement applicable plant commitments to (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed (See subsection 3.1.2.2.16)	R-25 R-35 R-76
28	PWR	Stainless steel and nickel alloy reactor vessel internals (except Westinghouse baffle former bolts)	Cracking due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Water Chemistry and FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.17)	R-112 R-118 R-133 R-150 R-162 R-167 R-186 R-194 R-203 R-210
29	PWR	Westinghouse stainless steel baffle former bolts and screws exposed to high fluence (>1 x 10e21 n/cm2, E >0.1 MeV)	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.18.1)	R-125

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
30	PWR	Stainless steel, cast austenitic stainless steel PH stainless steel forging reactor vessel internals components (except Westinghouse baffle former bolts)	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Water Chemistry and FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.18.2)	R-106 R-109 R-116 R-120 R-123 R-130 R-138 R-143 R-146 R-149 R-155 R-159 R-166 R-172 R-173 R-175 R-176 R-180 R-181 R-185 R-193 R-202 R-209 R-214
31	BWR	Stainless steel and nickel alloy penetrations for control rod drive stub tubes instrumentation, jet pump instrument, standby liquid control, flux monitor, and drain line exposed to reactor coolant	Cracking due to stress corrosion cracking, Intergranular stress corrosion cracking, cyclic loading	BWR penetrations and water chemistry	No	R-69

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
32	BWR	Stainless steel, cast austenitic stainless steel, and nickel alloy piping, piping components, and piping elements greater than or equal to 4 NPS; nozzle safe ends and associated welds	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	BWR Stress Corrosion Cracking and Water Chemistry	No	R-20 R-21 R-22 R-68
33	BWR	Stainless steel, nickel alloy vessel shell attachment welds exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	BWR vessel ID attachment welds and water chemistry	No	R-64
34	BWR	Stainless steel fuel supports and control rod drive assemblies control rod drive housing exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	BWR vessel internals and water chemistry	No	R-104
35	BWR	Stainless steel, cast austenitic stainless steel, nickel alloy core shroud, core plate, core plate bolts, support structure, top guide, core spray lines, spargers, jet pump assemblies, control rod drive housing, nuclear instrumentation guide tubes	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	BWR vessel internals and water chemistry	No	R-100 R-105 R-92 R-93 R-96 R-97 R-98 R-99
36	BWR	Steel (with or without stainless steel cladding) control rod drive return line nozzles exposed to reactor coolant	Cracking due to cyclic loading	CRD return line nozzle	No	R-66
37	BWR	Steel (with or without stainless steel cladding) feedwater nozzles exposed to reactor coolant	Cracking due to cyclic loading	Feedwater nozzle	No	R-65
38	BWR	Steel piping, piping components, and piping elements exposed to reactor coolant	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	R-23

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
39	BWR	Nickel alloy core shroud and core plate access hole cover (welded and mechanical covers)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Inservice inspection and water chemistry	No	R-95
40	BWR	Steel top head enclosure (without cladding) top head nozzles (vent, top head spray or RCIC, and spare) exposed to reactor coolant	Loss of material due to general, pitting and crevice corrosion	Inservice inspection and water chemistry	No	R-59
41	BWR	Nickel alloy core shroud and core plate access hole cover (welded and mechanical covers)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Inservice inspection, water chemistry, and augmented inspection of the access hole cover welds	No	R-94
42	BWR	High-strength low alloy steel top head closure studs and nuts exposed to air with reactor coolant leakage	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	Reactor head closure studs	No	R-60
43	BWR	Jet pump assembly castings; orificed fuel support	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	R-101 R-103

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
44	BWR/ PWR	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high-pressure and high-temperature systems	Cracking due to stress corrosion cracking, loss of material due to wear, loss of preload due to stress relaxation	Bolting Integrity	No	R-10 R-11 R-12 R-26 R-27 R-29 R-32 R-78 R-79 R-80
45	BWR/ PWR	Copper alloy piping, piping components, and piping elements exposed to closed cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Closed-cycle cooling water system and One-Time Inspection	No	RP-11
46	BWR/ PWR	Steel piping, piping components, and piping elements exposed to closed cycle cooling water	Loss of material due to general, pitting and crevice corrosion	Closed-cycle cooling water system and One-Time Inspection	No	RP-10
47	BWR/ PWR	Cast austenitic stainless steel Class 1 pump casings, and valve bodies and bonnets exposed to reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection. Thermal aging susceptibility screening is not necessary, inservice inspection requirements are sufficient for managing these aging effects. ASME Code Case N-481 also provides an alternative for pump casings.	No	R-08
48	BWR/ PWR	Copper alloy >15% Zn piping, piping components, and piping elements exposed to closed cycle cooling water	Loss of material due to selective leaching	Selective Leaching of Materials	No	RP-12
49	BWR/ PWR	Cast austenitic stainless steel piping and CRD pressure housings	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	R-52 R-77

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
50	PWR	Steel reactor coolant pressure boundary external surfaces	Loss of material due to boric acid corrosion	Boric acid corrosion	No	R-17
51	PWR	Steel steam generator steam nozzle and safe end, feedwater nozzle and safe end, AFW nozzles and safe ends exposed to secondary feedwater/steam	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	R-37 R-38
52	PWR	Stainless steel and nickel alloy reactor vessel internals exposed to reactor coolant	Loss of material due to wear	Inservice inspection	No	R-115 R-142 R-148 R-152 R-156 R-170 R-179 R-190 R-208 R-87
53	PWR	Stainless steel, steel pressurizer integral support exposed to air with metal temperature up to 288°C (550°F)	Cracking due to cyclic loading	Inservice inspection	No	R-19
54	PWR	Stainless steel, steel with stainless steel cladding reactor coolant system cold leg, hot leg, surge line, and spray line piping and fittings exposed to reactor coolant	Cracking due to cyclic loading	Inservice inspection	No	R-56
55	PWR	Steel steam generator secondary manways and handholds	Loss of material due to erosion	Inservice inspection	No	R-31
56	PWR	Stainless steel flux thimble tubes	Loss of material due to wear	Inservice inspection and recommendations of NRC IEB 88-09	No	R-145

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
57	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy and associated welds and buttering nozzle safe ends inlet outlet safety injection exposed to reactor coolant	Cracking due to stress corrosion cracking, primary water stress corrosion cracking	Inservice inspection and water chemistry	No	R-83
58	PWR	Stainless steel Class 1 piping, fittings, primary nozzles, safe ends, manways, flanges, CRD housing; pressurizer heater sheaths, sleeves, heater bundle diaphragm plate; pressurizer relief tank components; steam generator divider plate	Cracking due to stress corrosion cracking	Inservice inspection and water chemistry	No	R-07 R-14 R-217 R-30 RP-17
59	PWR	Steel with stainless steel or nickel alloy cladding; or stainless steel pressurizer components exposed to reactor coolant	Cracking due to cyclic loading	Inservice inspection and water chemistry	No	R-58
60	PWR	High-strength low alloy steel closure head stud assembly exposed to air with reactor coolant leakage	Cracking due to stress corrosion cracking; loss of material due to wear	Reactor head closure studs	No	R-71 R-72
61	PWR	Chrome plated nickel alloy, and stainless steel, steam generator anti-vibration bars exposed to secondary feedwater/ steam	Cracking due to stress corrosion cracking, loss of material due to crevice corrosion and fretting	Steam generator tubing integrity and water chemistry	No	RP-14 RP-15
62	PWR	Nickel alloy steam generator tubes, repair sleeves, and tube plugs exposed to reactor coolant	Cracking due to primary water stress corrosion cracking	Steam generator tubing integrity and water chemistry	No	R-40 R-44

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
63	PWR	Nickel alloy steam generator tubes and sleeves exposed to secondary feedwater/ steam	Cracking due to OD stress corrosion cracking and intergranular attack, loss of material due to fretting and wear	Steam generator tubing integrity and water chemistry	No	R-47 R-48 R-49
64	PWR	Nickel alloy steam generator tubes and sleeves exposed to phosphate chemistry in secondary feedwater/ steam	Loss of material due to wastage and pitting corrosion	Steam generator tubing integrity and water chemistry	No	R-50
65	PWR	Steel steam generator tube support plate, tube bundle wrapper exposed to secondary feedwater/steam	Loss of material due to erosion, general, pitting, and crevice corrosion, ligament cracking due to corrosion	Steam generator tubing integrity and water chemistry	No	R-42 RP-16
66	PWR	Nickel alloy steam generator tubes exposed to secondary feedwater/ steam	Denting due to corrosion of steel tube support plate	Steam generator tubing integrity and water chemistry and, for plants that could experience denting at the upper support plates, evaluate potential for rapidly propagating cracks and then develop and take corrective actions consistent with Bulletin 88-02.	No	R-43

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

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
67	PWR	Reactor vessel internals (CASS)	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	R-111 R-140 R-153 R-171 R-183 R-191 R-206
68	PWR	Cast austenitic stainless steel, steel with stainless steel cladding Class 1 pump casings and valve bodies exposed to reactor coolant	Cracking due to stress corrosion cracking	Water chemistry (and Inservice Inspection for components that do not meet the material guidelines of NUREG-0313).	No	R-09
69	PWR	Nickel alloy steam generator components such as, secondary side nozzles (vent, drain, and instrumentation) exposed to secondary feedwater/ steam	Cracking due to stress corrosion cracking	Water Chemistry and One-Time Inspection or Inservice Inspection.	No	R-36
70	BWR/ PWR	Stainless steel piping, piping components, and piping elements exposed to air with borated water leakage or gas	None	None	NA - No AEM or AMP	RP-07
71	BWR/ PWR	Stainless steel, cast austenitic stainless steel, and nickel alloy piping, piping components, and piping elements exposed to air – indoor uncontrolled (external)	None	None	NA - No AEM or AMP	RP-02 RP-03 RP-04
72	BWR/ PWR	Steel and stainless steel piping, piping components, and piping elements in concrete	None	None	NA - No AEM or AMP	RP-01 RP-06

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						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
73	PWR	Stainless steel piping, piping components, and piping elements exposed to air with borated water leakage or gas	None	None	NA - No AEM or AMP	RP-05

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, 2001 edition through the 2002 and 2003 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 and 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.

This Page Intentionally Left Blank