3.1 Aging Management of Reactor Vessel, Internals, and Reactor Coolant System

This section of the SER documents the staff's review of the applicant's aging management review (AMR) results for the reactor vessel, internals, and reactor coolant system components and component groups associated with the following systems:

- reactor vessel and control element drive mechanism pressure boundary
- reactor vessel internals
- Class 1 piping, valves, and reactor coolant pumps
- pressurizer
- steam generators (SGs)

3.1.1 Summary of Technical Information in the Application

In Section 3.1 of the LRA, the applicant provided the AMR results for the reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types listed in Tables 2.3.1-1 through 2.3.1-5 of the LRA. The applicant also listed the materials, environments, aging effects requiring management, and aging management programs associated with each system.

In Table 3.1.1, "Summary of Aging Management Programs for the Reactor Coolant System Evaluated in Chapter IV of NUREG-1801," of the LRA, the applicant provided a summary comparison of its AMRs with the AMRs evaluated in the GALL Report for the reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types. In Section 3.1.2.2 of the LRA, the applicant provided information concerning Table 3.1.1 components for which further evaluation is recommended by the GALL Report.

3.1.2 Staff Evaluation

The staff reviewed Section 3.1 of the LRA to understand the applicant's review process and to determine whether the applicant provided sufficient information to demonstrate that the effects of aging for the reactor vessel, internals, reactor coolant system, pressurizer, and SG components that are within the scope of license renewal and subject to an AMR will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

The staff performed an audit and review to confirm the applicant's claim that certain identified AMRs are consistent with the staff-approved AMRs in the GALL Report. The staff did not repeat its review of the matters described in the GALL Report. However, the staff did verify that the material presented in the LRA was applicable and that the applicant had identified the appropriate GALL AMRs. The staff's audit findings are summarized in Section 3.1.2.1 of this SER.

The staff also audited and reviewed those AMRs that are consistent with the GALL Report and for which further evaluation is recommended. The staff verified that the applicant's further evaluations were consistent with the acceptance criteria in Section 3.1.3.2 of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR). The staff's audit findings are summarized in Section 3.1.2.2 of this SER.

The staff conducted a technical review of the remaining AMRs that are not consistent with the GALL Report. The review included evaluating whether all plausible aging effects were identified and whether the aging effects listed were appropriate for the combination of materials and environments specified. The staff's review findings are summarized in Section 3.1.2.3 of this SER.

Finally, the staff reviewed the AMP summary descriptions in the UFSAR Supplement to ensure that they provide an adequate description of the programs credited with managing or monitoring aging for the reactor vessel, internals, and reactor coolant system components and component groups.

The staff's review of the reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types followed one of several approaches. One approach, documented in Section 3.1.2.1 of this SER, involves the staff's review of the AMR results for the reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types that the applicant indicated are consistent with the GALL Report and do not require further evaluation. Another approach, documented in Section 3.1.2.2, involves the staff's review of the AMR results for the reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types that the applicant indicated are consistent with the GALL Report and for which further evaluation is recommended. A third approach, documented in Section 3.1.2.3, involves the staff's review of the AMR results reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types that the applicant indicated are not consistent with the GALL Report or are not addressed in the GALL Report. The staff's review of AMPs that are credited to manage or monitor aging effects of the steam and power conversion system components is documented in Section 3.0.3 of this SER.

3.1.2.1 Aging Management Evaluations that are Consistent with the GALL Report, for Which No Further Evaluation is Required

Summary of Technical Information in the Application

In Section 3.1.2.1 of the LRA, the applicant identified the materials, environments, and aging effects requiring management. The applicant identified the following programs that manage the aging effects related to the reactor vessel, internals, reactor coolant system, pressurizer, and SG components:

- Reactor Vessel Integrity Program
- Inservice Inspection Program
- Water Chemistry Control Program
- Boric Acid Corrosion Prevention Program
- Alloy 600 Aging Management Program
- Reactor Vessel Head Penetration Program
- Bolting and Torquing Activities Program
- Reactor Vessel Internals Cast Austenitic Stainless Steel Components Program
- Reactor Vessel Internals Stainless Steel Plates, Forgings, Welds, and Bolting Program
- Cast Austenitic Stainless Steel Evaluation Program
- Pressurizer Examinations Program

- Steam Generator Integrity Program
- Flow-accelerated Corrosion Program

Staff Evaluation

In Tables 3.1.2-1 through 3.1.2-5 of the LRA, the applicant provided a summary of AMRs for the reactor vessel, internals, reactor coolant system, pressurizer, and SGs, and identified which AMRs it considered to be consistent with the GALL Report.

For component groups evaluated in the GALL Report for which the applicant has claimed consistency with the GALL Report, and for which the GALL Report does not recommend further evaluation, the staff performed an audit and review to determine whether the plant-specific components contained in these GALL Report component groups were bounded by the GALL Report evaluation.

The applicant provided a note for each AMR line item. The notes described how the information in the tables aligns with the information in the GALL Report. The staff audited those AMRs with Notes A through E, which indicated the AMR was consistent with the GALL Report.

Note A indicated that the AMR line item is consistent with the GALL Report for component, material, environment, and aging effect. In addition, the AMP is consistent with the AMP identified in the GALL Report. The staff audited these line items to verify consistency with the GALL Report and the validity of the AMR for the site-specific conditions.

Note B indicated that the AMR line item is consistent with the GALL Report for component, material, environment, and aging effect. In addition, the AMP takes some exceptions to the AMP identified in the GALL Report. The staff audited these line items to verify consistency with the GALL Report. The staff verified that the identified exceptions to the GALL AMPs had been reviewed and accepted by the staff. The staff also determined whether the AMP identified by the applicant was consistent with the AMP identified in the GALL Report and whether the AMR was valid for the site-specific conditions.

Note C indicated that the component for the AMR line item is different, but consistent with the GALL Report for material, environment, and aging effect. In addition, the AMP is consistent with the AMP identified by the GALL Report. This note indicates that the applicant was unable to find a listing of some system components in the GALL Report. However, the applicant identified a different component in the GALL Report that had the same material, environment, aging effect, and AMP as the component that was under review. The staff audited these line items to verify consistency with the GALL Report. The staff also determined whether the AMR line item of the different component was applicable to the component under review and whether the AMR was valid for the site-specific conditions.

Note D indicated that the component for the AMR line item is different, but consistent with the GALL Report for material, environment, and aging effect. In addition, the AMP takes some exceptions to the AMP identified in the GALL Report. The staff audited these line items to verify consistency with the GALL Report. The staff verified whether the AMR line item of the different component was applicable to the component under review. The staff verified whether the identified exceptions to the GALL AMPs had been reviewed and accepted by the staff. The

staff also determined whether the AMP identified by the applicant was consistent with the AMP identified in the GALL Report and whether the AMR was valid for the site-specific conditions.

Note E indicated that the AMR line item is consistent with the GALL Report for material, environment, and aging effect, but a different aging management program is credited. The staff audited these line items to verify consistency with the GALL Report. The staff also determined whether the identified AMP would manage the aging effect consistent with the AMP identified by the GALL Report and whether the AMR was valid for the site-specific conditions.

The staff conducted an audit and review of the information provided in the LRA and program bases documents, which are available at the applicant's engineering office. On the basis of its audit and review, the staff finds that the AMR results, which the applicant claimed to be consistent with the GALL Report, are consistent with the AMRs in the GALL Report. Therefore, the staff finds that the applicable aging effects were identified and are appropriate for the combination of materials and environments listed.

On the basis of its audit and review, the staff concludes that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

Staff RAIs Pertaining to Recent Operating Experience and Emerging Issues

Because the GALL Report and SRP-LR were issued in July 2001, these documents do not reflect the most current recommendations for managing certain aging effects that have been the subject of recent operating experience or the topic of an emerging issue. As a result, the staff issued RAIs to determine how the applicant proposed to address these items for license renewal. The applicant's responses to these RAIs, and the staff's evaluations of the responses, are documented as follows.

< Evaluation To Be Provided by DE/EMEB >

Conclusion

The staff has verified the applicant's claim of consistency with the GALL Report. The staff also has reviewed information pertaining to the applicant's consideration of recent operating experience and proposals for managing associated aging effects. On the basis of its review, the staff finds that the AMR results, which the applicant claimed to be consistent with the GALL Report, are consistent with the AMRs in the GALL Report. Therefore, the staff finds that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that their intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2 Aging Management Evaluations that are Consistent with the GALL Report, for Which Further Evaluation is Recommended

Summary of Technical Information in the Application

In Section 3.1.2.2 of the LRA, the applicant provided further evaluation of aging management as recommended by the GALL Report for reactor vessel, internals, and reactor coolant system components. The applicant provided information concerning how it will manage the following aging effects:

- cumulative fatigue damage
- loss of material due to pitting and crevice corrosion
- loss of fracture toughness due to neutron irradiation embrittlement
- crack initiation and growth due to thermal and mechanical loading or stress corrosion cracking
- crack growth due to cyclic loading
- changes in dimension due to void swelling
- crack initiation and growth due to stress corrosion cracking or primary water stress corrosion cracking
- crack initiation and growth due to stress corrosion cracking or irradiation-assisted stress corrosion cracking
- loss of preload due to stress relaxation
- loss of section thickness due to erosion
- crack initiation and growth due to PWSCC, ODSCC, or integranular attack or loss of material due to wastage and pitting corrosion or loss of section thickness due to fretting and wear or denting due to corrosion of carbon steel tube support plate
- loss of section thickness due to flow-accelerated corrosion
- ligament cracking due to corrosion
- loss of material due to flow-accelerated corrosion
- quality assurance for aging management of non-safety-related components

Staff Evaluation

For component groups evaluated in the GALL Report for which the applicant has claimed consistency with the GALL Report, and for which the GALL Report recommends further evaluation, the staff reviewed the applicant's evaluation to determine whether it adequately addressed the issues that were further evaluated. In addition, the staff reviewed the applicant's further evaluations against the criteria contained in Section 3.1.2.2 of the SRP-LR. Details of the staff's audit and review are documented in the staff's audit and review report.

The GALL Report indicates that further evaluation should be performed for the aging effects described in the following sections of this SER.

3.1.2.2.1 Cumulative Fatigue Damage

< To be reviewed by DE >

3.1.2.2.2 Loss of Material Due to Pitting and Crevice Corrosion

In Section 3.1.2.2.2 of the LRA, the applicant addressed loss of material of SG assemblies due to pitting and crevice corrosion.

SRP-LR Section 3.1.2.2.2 states that loss of material due to pitting and crevice corrosion could

occur in the SG shell assembly. The existing program relies on control of water chemistry to mitigate corrosion and ISI to detect loss of material. NRC IN 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," states that if general corrosion pitting of the shell exists, the existing program may not be sufficient. In that case, the GALL Report recommends augmented inspections to manage the aging effect.

The AMPs recommended by the GALL Report for managing the aging of SG assemblies due to pitting and crevice corrosion are ASME Section XI inservice inspection, Subsections IWB, IWC, and IWD (XI.M1) program to detect loss of material and the water chemistry (XI.M2) program to mitigate corrosion. The GALL Report recommends a plant-specific program to conduct augmented inspections.

The applicant credited the inservice inspection program (AMP B.1.14) and the primary and secondary water chemistry control program (AMP B.1.30.3) for managing loss of material due to pitting and crevice corrosion on the internal surfaces of the SG shell. The staff reviewed the inservice inspection program and the primary and secondary water chemistry control program and its evaluation of these programs is documented in Sections 3.0.3.3.5 and 3.0.3.1 of this SER, respectively.

The staff reviewed IN 90-04, which identifies the need to augment inspections beyond the requirements of ASME Section XI if general corrosion pitting of the SG shell is known to exist in order to differentiate isolated cracks for inherent geometric conditions. The applicant replaced the its SGs in 2000. The staff reviewed operating experience which indicated that no pitting corrosion of the SG shell has been detected to date, and that water chemistry has been maintained for these new SGs per EPRI guidelines. The staff finds that the augmented inspections recommended by NRC IN 90-04, as referenced in the SRP-LR, do not currently apply to the the applicant's SGs.

Since pitting corrosion has not been detected on the SG shell since installation, the staff finds that augmented inspections are not required and that the current water chemistry control and inservice inspection programs are adequate to manage aging.

The staff finds that the applicant has demonstrated that the effect of aging for loss of material due to pitting and crevice corrosion will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation.

3.1.2.2.3 Loss of Fracture Toughness Due to Neutron Irradiation Embrittlement

In Section 3.1.2.2.3 of the LRA, the applicant addressed (1) loss of fracture toughness due to neutron irradiation embrittlement for ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 at the end of the license renewal term, (2) loss of fracture toughness due to irradiation embrittlement of the reactor vessel beltline materials. In addition, the applicant stated that (3) the ANO-2 reactor vessel internals do not include baffle/former bolts. The core shroud plates are joined in a welded configuration and the discussion in this paragraph of NUREG-1800 is not applicable to ANO-2.

Section 3.1.2.2.3 of the SRP-LR states that certain aspects of neutron irradiation embrittlement are TLAAs as defined in 10 CFR 54.3 and that TLAAs are required to be evaluated in

accordance with 10 CFR 54.2.(c)(1). Second, Section 3.1.2.2.3 of the SRP-LR states that loss of fracture toughness due to neutron irradiation embrittlement could occur in the reactor vessel. A reactor vessel materials surveillance program monitors neutron irradiation embrittlement of the reactor vessel. Reactor vessel surveillance programs are plant-specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant is required to submit its proposed withdrawal schedule for approval prior to implementation. Finally, the SRP-LR Section 3.1.2.2.3 statement that loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur in Westinghouse and B&W baffle/former bolts is not applicable to ANO-2 because ANO-2 reactor vessel internals do not include baffle/former bolts.

The AMP recommended by the GALL Report for managing loss of fracture toughness due to neutron irradiation embrittlement in the reactor vessel is XI.M31, “Reactor Vessel Surveillance,” which complies with the requirements of 10 CFR Part 50, Appendices G and H, and 10 CFR Part 50.61.

Loss of fracture toughness due to neutron irradiation embrittlement for ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 at the end of the license renewal term is a TLAA, described in Section 4.2 of the LRA. **< To be reviewed by DE >**

The applicant stated in the LRA that loss of fracture toughness due to irradiation embrittlement of the reactor vessel beltline materials is managed by the reactor vessel integrity program (AMP B.1.21). This program includes a plant-specific material surveillance program which monitors the effect of operational fluence levels on material specimens (surveillance capsules) located in the reactor vessel during power operations. These surveillance capsules are periodically withdrawn and analyzed. The applicant has an approved withdrawal schedule for surveillance capsules and has, as part of their reactor vessel integrity program, evaluated AMR results involving managing the reduction of fracture toughness of reactor vessel beltline materials, as recommended in the GALL Report. The applicant committed in its LRA to revise the specimen capsule withdrawal schedule and test a standby capsule to cover the peak fluence expected through the end of the period of extended operation. The staff reviewed the reactor vessel integrity program and its evaluation is documented in Section 3.0.3.2.6 of this SER. **< To be reviewed/verified by DE >**

The staff finds that the applicant's AMR results are consistent with the GALL Report and that the applicant has demonstrated that programs to manage the effects of aging will be adequate to maintain the intended functions consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2.4 Crack Initiation and Growth Due to Thermal and Mechanical Loading or Stress Corrosion Cracking

In Section 3.1.2.2.4 of the LRA, the applicant addressed the potential crack initiation and growth due to thermal and mechanical loading or stress corrosion cracking (SCC) (including intergranular SCC) that could occur in small-bore RCS and connected system piping less than 4-inch nominal pipe size (NPS 4).

Section 3.1.2.2.4 of the SRP-LR states that 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 applicant should verify that service-induced weld cracking is not occurring in small-bore piping less than NPS 4. 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. Per ASME Section XI, 1995 Edition, Examination Category B-J or B-F, small bore piping, defined as piping less than NPS 4, does not receive volumetric inspection.

The AMPs recommended by the GALL Report are XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," to detect loss of material and XI.M2, "Water Chemistry," to mitigate SCC. The GALL Report recommends GALL AMP XI.M32, "One-Time Inspection," as an acceptable verification method to ensure that cracking is not occurring in small bore piping.

The applicant credited the inservice inspection program (AMP B.1.14) and the primary and secondary water chemistry control program (AMP B.1.30.3) to mitigate cracking of reactor coolant piping. The staff's review of these programs is documented in Sections 3.0.3.3.5 and 3.0.3.1 of this SER, respectively.

To address the GALL Report recommendation of a plant-specific destructive examination or an NDE for inspection of inside surfaces of small bore piping, the applicant stated, in LRA Section 3.1.2.2.4 and Table 3.1.1-7, that it has implemented a risk-informed methodology at ANO-2 to select, for small bore RCS and connected systems piping, RCS piping welds for inspection. The applicant stated, in LRA Section 3.1.2.2.4, that the current inspection methods as described in the inservice inspection program appropriately address cracking of small bore piping systems less than four inch nominal pipe schedule (NPS 4) and greater than 1-inch (NPS 1). The staff finds that this methodology appropriately addresses cracking of small bore piping greater than NPS 1, and the risk-informed methodology adequately manages cracking initiation and growth aging mechanisms during the period of extended operation.

In Section 3.1.2.2.4 of the LRA, the applicant stated that, for NPS 1 RCS piping and smaller, the piping is austenitic stainless steel and is not within the scope of the risk-informed selection of piping welds for inspection. The applicant further stated that volumetric examinations of NPS 1 RCS piping and smaller are not effective, and the applicant performs system leakage testing, in accordance with ASME Section XI, as the preferred alternative to inspection of the inside surfaces of small bore piping NPS 1 and smaller.

In discussions with the applicant, the staff asked the applicant to clarify how the alternative of system leakage testing for NPS 1 RCS piping and smaller will adequately manage aging of small bore piping and to provide the technical basis for not including piping NPS 1 and smaller in the sample inspections from the risk-informed selections.

In its response, the applicant stated that operating experience has confirmed that leakage from NPS 1 and smaller piping is readily detected and corrected prior to loss of system function. Additionally, the applicant stated that it had implemented a program to investigate the potential

for cracking of welded joints in RCS piping less than or equal to NPS 1 since the discovery of a cracked weld in an ANO-1 RCS drain line in 1989. Additionally, the applicant stated that the risk (based on probability and consequences) of failure of the 1-inch and smaller piping is less than the risk of failures of locations selected for inspection in the small-bore piping inspection program.

On the basis of the applicant's response and its review, the staff finds that visual inspection of NPS 1 and smaller RCS piping using systems leakage testing, in conjunction with volumetric examinations of NPS 1 to NPS 4 RCS piping of the same material and environment, adequately manages the effects of crack initiation and growth due to thermal and mechanical loading or stress corrosion cracking prior to loss of intended function. This approach is consistent with that for ANO-1 which was evaluated and approved by the staff in NUREG-1743, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1.

The staff finds that the applicant has demonstrated that crack initiation and growth due to thermal and mechanical loading or SCC on small-bore RCS and connected systems piping will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation.

3.1.2.2.5 Crack Growth Due to Cyclic Loading

As stated in the SRP-LR, fatigue is a time-limited aging analysis (TLAA) as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff's review of the applicant's evaluation of this TLAA is documented in Section 4.3 of this SER. In performing this review, the staff followed the guidance in Section 4.3 of the SRP-LR.
< To be reviewed/verified by DE >

3.1.2.2.6 Changes in Dimension Due to Void Swelling

In Section 3.1.2.2.6 of the LRA, the applicant addressed changes in dimension due to void swelling that could occur in reactor internal components.

Section 3.1.2.2.6 of the SRP-LR states that the GALL Report recommends that changes in dimension due to void swelling in reactor internal components be evaluated to ensure that this aging effect is adequately managed. The GALL Report recommends that a plant-specific AMP be evaluated to manage the effects of changes in dimension due to void swelling and the loss of fracture toughness associated with swelling.

The applicant stated that the void swelling of reactor vessel internals is managed by the reactor vessel internals cast austenitic stainless steel (CASS) program (AMP B.1.22) and the reactor vessel internals stainless steel plates, welds, forgings, and bolting program (AMP B.1.23) using supplemental examinations or component-specific evaluations. The applicant has committed to further understanding of this aging effect through industry programs to provide additional bases for supplemental examinations or component-specific evaluations.

The staff evaluated the reactor vessel internals CASS program and the reactor vessel internals stainless steel plates, welds, forgings, and bolting program. The staff documented its results in Section 3.0.3.1 of this SER. These programs will be consistent with GALL AMPs XI.M13,

"Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)," and XI.M16, "PWR Vessel Internals," respectively.

The staff finds the applicant's approach for managing changes in dimension due to void swelling reasonable because the approach will be based on the guidelines developed by the ongoing industry activities related to void swelling. The applicant has committed to submitting both AMPs B.1.22 and B.1.23 to the staff for review and approval three years prior to the period of extended operation.

3.1.2.2.7 Crack Initiation and Growth Due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking

The staff reviewed Section 3.1.2.2.7 of the LRA against the criteria in SRP-LR Section 3.1.2.2.7, which recommends plant-specific programs to address these aging mechanisms.

In Section 3.1.2.2.7 of the LRA, the applicant addressed (1) crack initiation and growth due to SCC and primary water stress corrosion cracking (PWSCC) in the surge nozzle thermal sleeve, safety injection nozzle thermal sleeve, charging inlet nozzle thermal sleeve, resistance temperature detector nozzles, pressure measurement nozzle, sampling nozzle, and partial nozzle replacement. Reactor vessel items included in this grouping are the lower shell and bottom head cladding, surveillance capsule holders, core stabilizing lugs, core stop and support lugs, and the flow baffle and skirt. Steam generator items included in this grouping are the tube plate cladding, channel head divider plate, and primary nozzle closure rings; (2) crack initiation and growth due to SCC in the pressurizer surge line piping and fittings fabricated of CASS; and (3) crack initiation and growth due to PWSCC in nickel-based alloy material such as the pressurizer instrumentation nozzles, heater sheaths and sleeves, and thermal sleeves. ANO-2 pressurizer components included in this grouping are the instrument nozzles, X-1 and T-4 heater penetration nozzles and plugs, original heater sheath, heater sleeve, and end plugs.

Section 3.1.2.2.7 of the SRP-LR states that

- Crack initiation and growth due to SCC and PWSCC could occur in core support pads (or core guide lugs), instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the SG instruments and drains. The GALL Report recommends further evaluation to ensure that these aging effects are adequately managed. The GALL Report recommends that a plant-specific AMP be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC.
- Crack initiation and growth due to SCC could occur in CASS RCS piping and fittings and pressurizer surge line nozzle. The GALL Report recommends further evaluation of piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313.
- Crack initiation and growth due to PWSCC could occur in pressurizer instrumentation penetrations and heater sheaths and sleeves made of nickel alloys. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate PWSCC. However, the existing program should be augmented to manage the effects of SCC on the intended function of nickel-alloy components. The GALL Report recommends that the applicant provide a plant-specific AMP or participate in industry

programs to determine appropriate AMPs for PWSCC of the Alloy 182 weld.

The applicant credited the following plant-specific programs for each of the three SRP-LR criteria:

- Cracking of nickel-based alloy components due to PWSCC is managed by the Alloy 600 aging management program (AMP B.1.1) supplemented by the water chemistry control program and the inservice inspection program. Additionally, EPRI, through its material reliability program (MRP) and in conjunction with the PWR owners groups, is developing a strategic plan to manage and mitigate cracking of nickel-based alloy items. The applicant has stated that the guidance developed by the MRP will be used to identify critical locations for inspection and to augment existing ISI inspections, as appropriate. Since RCS pressure control using the pressurizer sprays is not an intended function of the pressurizer, the pressurizer spray assembly is not subject to aging management for ANO-2.
- Crack initiation and growth due to SCC at welded connections, including the pressurizer surge line and fittings, is managed by the water chemistry control program and the inservice inspection program.
- The programs credited for the management of PWSCC of these nickel-based alloy items are the Alloy 600 aging management program and the water chemistry control program, supplemented by the inservice inspection program. As described in Item 1 above, the applicant committed to participation in the Alloy 600 industry programs to identify critical locations for inspection and augment existing ISI, where appropriate.

The staff reviewed the plant-specific programs for these aging effects as follows:

- The staff's evaluation of the primary and secondary water chemistry control program (AMP B1.30.3) is documented in Section 3.0.3.1 of this SER.
- The staff's evaluation of the inservice inspection program (AMP B.1.14) is documented in Section 3.0.3.3.5 of this SER.

The staff's evaluation of the Alloy 600 aging management program (AMP B.1.1) is documented in Section 3.0.3.3.1 of this SER.

The staff finds that the applicant appropriately evaluated AMR results which address these aging mechanisms, as recommended in the GALL Report.

On the basis of its review, the staff finds that the applicant appropriately evaluated AMR results involving management of crack initiation and growth due to SCC or PWSCC, as recommended in the GALL Report.

3.1.2.2.8 Crack Initiation and Growth Due to Stress Corrosion Cracking or Irradiation-Assisted Stress Corrosion Cracking

In Section 3.1.2.2.8 of the LRA, the applicant stated that its reactor vessel internals do not include baffle/former bolts. The core shroud plates are joined in a welded configuration and that the discussion in this paragraph of NUREG-1800 is not applicable.

On the basis that the baffle/former bolts are not part of the design of reactor vessel internals, the staff finds that this aging effect is not applicable.

3.1.2.2.9 Loss of Preload Due to Stress Relaxation

In Section 3.1.2.2.9 of the LRA, the applicant stated that its reactor vessel internals do not include baffle/former bolts. The core shroud plates are joined in a welded configuration and that the discussion in this paragraph of NUREG-1800 is not applicable.

On the basis that the baffle/former bolts are not part of the design of reactor vessel internals, the staff finds that this aging effect is not applicable.

3.1.2.2.10 Loss of Section Thickness Due to Erosion

In Section 3.1.2.2.10 of the LRA, the applicant stated that its steam generators do not include impingement plates and that the discussion in this paragraph is not applicable.

Section 3.1.2.2.10 of the SRP-LR states that loss of section thickness due to erosion could occur in SG feedwater impingement plates and supports. The GALL Report recommends further evaluation of a plant-specific AMP to ensure that this aging effect is adequately managed.

On the basis that impingement plates are not part of the steam generator design, the staff finds that this aging effect is not applicable.

3.1.2.2.11 Crack Initiation and Growth Due to Primary Water Stress Corrosion Cracking, Outside Diameter Stress Corrosion Cracking, or Intergranular Attack or Loss of Material Due to Wastage and Pitting Corrosion or Loss of Section Thickness Due to Fretting and Wear or Denting Due to Corrosion of Carbon Steel Tube Support Plate

In Section 3.1.2.2.11 of the LRA, the applicant addressed crack initiation and growth due to PWSCC, outside diameter SCC (ODSCC,) or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion or deformation due to corrosion that could occur in nickel-based alloy components of the SG tubes and plugs.

Section 3.1.2.11 of the SRP-LR states that crack initiation and growth due to PWSCC, ODSCC, or IGA or loss of material due to wastage and pitting corrosion or deformation due to corrosion could occur in Alloy 600 components of the SG tubes, repair sleeves and plugs. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL Report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, should be developed to ensure that this aging effect is adequately managed.

The SRP-LR also states that crack initiation and growth due to PWSCC, ODSCC or IGA or loss of material due to wastage and pitting corrosion or deformation due to corrosion could occur in nickel-based alloy components of the SG tubes and plugs.

To manage the effects of aging, the applicant credited the SG integrity program (AMP B.1.25) supplemented by the primary and secondary water chemistry control program (AMP B.1.30.3) and the inservice inspection program (AMP B.1.14).

The staff's evaluation of the SG integrity program is documented in Section 3.0.3.1 of this SER. The staff evaluated the primary and secondary water chemistry control and the inservice inspection program and its evaluations are documented in Sections 3.0.3.1 and 3.0.3.3.5 of this SER, respectively. For general and pitting corrosion and for the assessment of tube integrity and plugging or repair criteria of flawed tubes, the SG integrity program acceptance criteria are in accordance with NEI 97-06 guidelines.

On the basis of its review of the primary and secondary water chemistry control program and the inservice inspection program, the staff finds that the applicant appropriately evaluated AMR results involving plant-specific programs to address these aging mechanisms, as recommended in the GALL Report.

3.1.2.2.12 Loss of Section Thickness Due to Flow-Accelerated Corrosion

In Section 3.1.2.2.12 of the LRA, the applicant states that its steam generators do not include carbon steel tube support lattice bars. Therefore, loss of section thickness of these bars is not an applicable aging effect.

On the basis that carbon steel tube support lattice bars are not part of the SG design, the staff finds that this aging effect is not applicable.

3.1.2.2.13 Ligament Cracking Due to Corrosion

In Section 3.1.2.2.13 of the LRA, the applicant states that the steam generators have stainless steel tube support plates. Therefore, ligament cracking due to corrosion is not an applicable aging effect.

On the basis that carbon steel components are not part of the SG tube support plate design, the staff finds that this aging effect is not applicable to ANO-2.

3.1.2.2.14 Loss of Material Due to Flow-Accelerated Corrosion

In Section 3.1.2.2.14 of the LRA, the applicant stated that the discussion in this paragraph of NUREG-1800 is applicable to CE System 80 steam generators only, whereas it has Westinghouse Delta 109 steam generators.

On the basis that CE System 80 SGs are not part of the ANO-2 SGs design, the staff finds that ANO-2 components are not subject to this aging effect and that this aging effect is not applicable to ANO-2.

3.1.2.2.15 Quality Assurance for Aging Management of Non-Safety-Related Components

< Evaluation To Be Provided by NRR DIPM >

Conclusion

On the basis of its review, for component groups evaluated in the GALL Report for which the applicant has claimed consistency with the GALL Report, and for which the GALL Report recommends further evaluation, the staff determines that the applicant adequately addressed the issues that were further evaluated. In addition, the staff reviewed the applicant's further evaluations against the criteria contained in the SRP-LR. Since the applicant's AMR results are otherwise consistent with the GALL Report, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.3 AMR Results that are Not Consistent with the GALL Report or Not Addressed in the GALL Report

Summary of Technical Information in the Application

In Tables 3.1.2-1 through 3.1.2-5 of the LRA, the staff reviewed additional details of the results of the AMRs for material, environment, aging effect requiring management, and AMP combinations that are not consistent with the GALL Report.

In Tables 3.1.2-1 through 3.1.2-5, the applicant indicated, via Notes F through J, that neither the identified component nor the material and environment combination is evaluated in the GALL Report and provided information concerning how the aging effect will be managed.

Note F indicated that the material is not in the GALL Report for the identified component.

Note G indicated that the environment is not in the GALL Report for the identified component and material.

Note H indicated that the aging effect is not in the GALL Report for component, material, and environment combination.

Note I indicated that the aging effect in the GALL Report for the identified component, material, and environment combination is not applicable

Note J indicated that neither the identified component nor the material and environment combination is evaluated in the GALL Report.

Staff Evaluation

For component type, material and environment combination that are not evaluated in the GALL Report, the staff reviewed the applicant's evaluation to determine whether the applicant had demonstrated that the effects of aging will be adequately managed so that the intended function will be maintained consistent with the CLB during the period of extended operation.

The staff evaluation is discussed below.

3.1.2.3.1 Reactor Vessel and CEDM Pressure Boundaries Summary of Aging Management - Table 3.1.2-1

The staff reviewed Table 3.1.2-1 of the LRA, which summarized the results of AMR evaluations in the SRP-LR for the reactor vessel and control element drive mechanism (CEDM) pressure boundary component groups.

The applicant proposed to manage loss of material for the following stainless steel, nickel-based alloy, and low alloy steel clad with stainless steel and nickel-based alloy component types of the reactor vessel and CEDM pressure boundary system - core stop lugs, flow skirt, and surveillance capsule holders; penetrations for the CEDM motor housing, CEDM upper pressure housing, CEDM ball seal housing, CEDM upper pressure housing upper fitting, CEDM motor housing upper and lower end fittings, CEDM upper pressure housing lower fitting, CEDM nozzle, ICI nozzle tubes, ICI flange adapter/seal plate, reactor vessel vent pipe, and reactor vessel vent pipe flange; reactor vessel shell and nozzles for the bottom head (torus and dome), upper shell, closure head dome (torus and dome), intermediate shell, lower shell, and primary inlet/outlet nozzle safe ends - exposed internally to treated, borated water using the primary and secondary water chemistry control program (AMP B.1.30.3). The staff's evaluation of the primary and secondary water chemistry control program is documented in Section 3.0.3.1 of this SER. The staff concludes that the primary and secondary water chemistry control program credited by the applicant for this line item is adequate.

For each of these same component and material combinations in Table 3.1.2-1, the applicant is also managing cracking using the water chemistry control program (AMP B.1.14), inservice inspection - inservice inspection program (AMP B.1.14), and a plant-specific program such as Alloy 600 aging management program (AMP B.1.1). The staff's evaluation of the inservice inspection - inservice inspection - IWB, IWC, IWD, and IWF program is documented in Section 3.0.3.3.5 of this SER. The staff concludes that the inservice inspection - inservice inspection program credited by the applicant for this line item is adequate. The staff reviewed the Alloy 600 aging management program and its evaluation is documented in Section 3.0.3.3.1 of this SER. **< To be reviewed/verified by DE >**. On the basis of the above discussion, the staff finds that the applicant manages cracking in a manner consistent with the GALL Report.

On the basis that management of cracking of stainless steel, nickel-based alloy and low alloy steel clad with stainless steel is being managed by the water chemistry control and inservice inspection programs, and the effects of pitting and crevice corrosion on stainless steel and nickel-based alloy components are not significant in chemically treated, borated water, the staff finds that management of loss of material using water chemistry control is adequate.

In the case of the stainless steel CEDM motor housing, upper-pressure housing and fitting, and ball seal housing as well as the CEDM nickel-alloy fittings, the staff asked the applicant to justify application of this position under the low-flow conditions that are expected. The staff reviewed a report (ML003748904) of maintenance activities that documented site-specific experience. This included a record of the visual inspection of materials in the same environment that had been operated under virtually identical conditions without observable loss of material, confirming the effectiveness of a water chemistry control program for management of this aging effect.

On the basis of industry and plant-specific operating experience, and the fact that the applicant manages the cracking aging effect of these same components, materials, and environment combinations using water chemistry control and inservice inspection programs, the staff finds that the use of a plant-specific water chemistry program to manage loss of material for stainless steel and nickel-based alloy components exposed to treated, borated water is acceptable.

The staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation.

<Additional Evaluation of LRA Notes F through J To Be Provided by DE/EMCB >

3.1.2.3.2 Reactor Vessel Internals Summary of Aging Management - Table 3.1.2-2

The staff reviewed Table 3.1.2-2 of the LRA, which summarized the results of AMR evaluations in the SRP-LR for the reactor vessel internals component groups.

The applicant proposed to manage loss of material for the following stainless steel component types of the reactor vessel internals system - control element assembly (CEA) shroud assembly components such as CEA instrument tube, CEA shroud adapter, CEA shroud support, positioning plate CEA shroud flow channel extension, and core shroud tube; core shroud assembly components such as core shroud plates, plates, ribs, intermediate plates, and core shroud guide lugs; and incore instrumentation (ICI) components such as guide tubes, ICI thimble support plate assembly, ICI support plate, grid, lifting support, lifting plate, columns, plates, funnel, pad, ring, nipple, hex bolt, spacer, threaded rod, hex jam nut, thimble support nut, and cap screws - exposed internally to treated, borated water using the primary and secondary water chemistry control program (AMP B.1.30.3). The staff's evaluation of the primary and secondary water chemistry control program is documented in Section 3.0.3.1 of this SER. The staff concludes that the primary and secondary water chemistry control program credited by the applicant for this line item is adequate.

For each of these same component and material combinations Table 3.1.2-2, the applicant is also managing cracking using the water chemistry control program, the inservice inspection - inservice inspection program (AMP B.1.14), and a plant-specific program such as reactor internals stainless steel program. The staff's evaluation of the inservice inspection - inservice inspection - IWB, IWC, IWD, and IWF program is documented in Section 3.0.3.3.5 of this SER. The staff concludes that the primary and secondary water chemistry control program credited by the applicant for this line item is adequate. The staff reviewed reactor vessel internals stainless steel plates, forgings, welds, and bolting (AMP B.1.23) and evaluation of this AMP is documented in Section 3.0.3.1 of this SER. On the basis of the above discussion, the staff finds that the applicant manages cracking in a manner consistent with the GALL Report.

On the basis that cracking of stainless steel is being managed by the water chemistry control and inservice inspection programs, and the effects of pitting and crevice corrosion on stainless steel components are not significant in chemically treated, borated water, the staff finds that management of loss of material using water chemistry control is adequate.

The GALL Report recommends a loose parts monitoring program to manage loss of

mechanical closure integrity for control element assembly shroud extension shaft guides, cylinders, and bases; shroud base; shroud flow channel; shroud flow channel cap; shroud shaft retention pin; shroud retention block; spanner nuts; shroud fasteners; guide tubes; incore instrumentation thimble support plate assembly; incore instrumentation support plate, grid, lifting support, lifting plate, column, plates, funnel; pad, ring, nipple, hex bolt, spacer; threaded rod, hex jam nut, thimble support nut, cap screws, reactor vessel internals.

The applicant proposed to manage this aging effect using the reactor vessel internals (stainless steel) program (AMP B.1.22) and inservice inspection - inservice inspection program (AMP B.1.14). The staff reviewed these programs and its evaluation is documented in Sections 3.0.3.1 and 3.0.3.3.5 of this SER, respectively. The staff concludes that the reactor vessel internals (stainless steel) program and the inservice inspection - inservice inspection program credited by the applicant for this line item is adequate.

On the basis that the reactor vessel internals programs detect aging effects prior to the loss of mechanical integrity of these components, the staff finds that the use in lieu of a loose parts monitoring program is acceptable. The staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation.

<Additional Evaluation of LRA Notes F through J To Be Provided by DE/EMCB >

3.1.2.3.3 Class 1 Piping, Valves, and Reactor Coolant Pumps Summary of Aging Management - Table 3.1.2-3

The staff reviewed Table 3.1.2-3 of the LRA, which summarized the results of AMR evaluations in the SRP-LR for the Class 1 piping, valves, and RCPs component groups. The staff finds that the programs proposed for management of aging effects for the component types in this system are consistent with the GALL Report.

<Additional Evaluation of LRA Notes F through J To Be Provided by DE/EMCB >

3.1.2.3.4 Pressurizer Summary of Aging Management - Table 3.1.2-4

The staff reviewed Table 3.1.2-4 of the LRA, which summarized the results of AMR evaluations in the SRP-LR for the pressurizer component groups.

During the review, the staff noted that the inservice inspection program (AMP B.1.14) had not been credited for managing the cracking of pressurizer safe ends. The staff requested the applicant to correct this discrepancy.

By letter dated March 24, 2004, the applicant committed to use the inservice inspection program to manage cracking of the pressurizer safe ends. This is now consistent with the GALL Report and acceptable to the staff.

For loss of material from the nickel-alloy pressurizer heater support plates and support brackets exposed to treated, borated water, the applicant credited the water chemistry control program. The staff's evaluation of the water chemistry control program is documented in Section 3.0.3.1

of this SER. The staff concludes that the water chemistry control program credited by the applicant for this line item is adequate.

On the basis of industry operating experience with this material and use of a water chemistry control program consistent with the GALL Report, the staff found this acceptable.

<Additional Evaluation of LRA Notes F through J To Be Provided by DE/EMCB >

3.1.2.3.5 Steam Generators Summary of Aging Management - Table 3.1.2-5

The staff reviewed Table 3.1.2-5 of the LRA, which summarized the results of AMR evaluations in the SRP-LR for the SG component groups.

For loss of material from the nickel-alloy SG tube plugs, U-tubes, divider plate, and tube plate exposed to treated, borated water, the applicant credited the primary and secondary water chemistry control program (AMP B.1.30.3). The staff's evaluation of the primary and secondary water chemistry control program is documented in Section 3.0.3.1 of this SER. The staff concludes that the primary and secondary water chemistry control program credited by the applicant for this line item is adequate.

On the basis of industry operating experience with this material and use of a water chemistry control program consistent with the GALL Report, the staff finds this acceptable.

The staff identified during its review of Table 3.1.2-5 of the LRA, that the SG inspection port diaphragms in the LRA had been associated with the wrong item from tables in the GALL Report. By letter dated March 24, 2004, the applicant revised the associated note for this component type.

The staff finds that management of cracking in nickel-based alloy exposed to treated water using water chemistry control program verified by inservice inspection is acceptable.

<Additional Evaluation of LRA Notes F through J To Be Provided by DE/EMCB >

Conclusion

On the basis of its review, the staff finds that the applicant appropriately evaluated AMR results involving material, environment, aging effect requiring management, and AMP combinations that are not evaluated in the GALL Report. The staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.3 Conclusion

The staff concluded that the applicant provided sufficient information to demonstrate that the effects of aging for the reactor vessel, internals, reactor coolant system, pressurizer, and SG components and component types that are within the scope of license renewal and subject to an AMR will be adequately managed so that the intended functions will be maintained

consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).