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NUCLEAR REGULATORY COMMISSION
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SALEM GENERATING STATION UNIT NOS. 1 AND 2 – AUTHORIZATION AND SAFETY EVALUATION FOR ALTERNATIVE REQUEST NO. SC-I4R-200 (EPID L-2020-LLR-0103)

LICENSEE INFORMATION

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Plant Name and Unit: Salem Generating Station Unit Nos. 1 and 2

Docket Nos.: 50-272 and 50-311

APPLICATION INFORMATION

Application Date: August 5, 2020

Application Agencywide Documents Access and Management System (ADAMS) Accession No.: ML20218A587

Supplement Date: April 12, 2021

Supplement ADAMS Accession No.: ML21102A024

Applicable Inservice Inspection (ISI) Program Interval: Remainder of the fourth 10-year ISI interval and through the following fifth 10-year ISI interval for Salem Generating Station (Salem) Unit Nos. 1 and 2. The fifth 10-year ISI interval is currently scheduled to end on December 31, 2030. The U.S. Nuclear Regulatory Commission (NRC) staff noted that the end date of the fifth 10-year ISI interval does not align with the end dates of the fifth decade of operation of Salem Unit Nos. 1 and 2 determined from the Salem Unit Nos. 1 and 2 operating licenses (ADAMS Accession Nos. ML052990140 and ML052990143, respectively). In the supplement dated April 12, 2021, the licensee clarified that due to extended outages on Salem Unit Nos. 1 and 2 in the mid-1990s, the 10-year ISI intervals are not in alignment with the dates of the Salem Unit Nos. 1 and 2 operating licenses. Additionally, the licensee clarified that the Salem Unit No. 2 fourth 10-year ISI interval was shortened to align with the end of the Salem Unit No. 1 fourth 10-year ISI interval and the containment inservice inspection 10-year intervals. The fifth 10-year ISI intervals for both Salem Unit Nos. 1 and 2 are scheduled to end on December 31, 2030, as documented in the ISI program plan.

Alternative Provision: The applicant requested an alternative under Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(z)(1).

ISI Requirements: For American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1 welds, the ISI requirements are those specified in Paragraph IWB-2411 of the ASME Code, Section XI, which requires the licensee to perform volumetric examinations of the following pressurizer (PZR) shell-to-head welds as specified in ASME Code, Section XI, Table IWB-2500-1 once every 10-Year ISI interval.

- Examination Category B-B, Item No. B2.11, PZR shell-to-head welds, circumferential
- Examination Category B-B, Item No. B2.12, PZR shell-to-head welds, longitudinal

Applicable Code Edition and Addenda: 2004 Edition of the ASME Code, Section XI, no addenda for the fourth 10-year ISI interval and 2013 Edition of the ASME Code, Section XI, no addenda for the fifth 10-year ISI interval.

Brief Description of the Proposed Alternative: In Section 6.0 of Attachment 1 to its submittal dated August 5, 2020, the licensee stated that the proposed alternative is to increase the ISI interval from the current ASME Code, Section XI, requirement of 10 years to 30 years for the following Salem Unit Nos. 1 and 2 PZR welds, as listed in Section 1.0 of Attachment 1 to the submittal.

Item No.	Weld ID	Component Description
B2.11	1-PZR-1	LOWER HEAD TO SHELL A, CIRC WELD
B2.11	1-PZR-21	SHELL J TO UPPER HEAD, CIRC WELD
B2.12	1-PZR-2	LONGITUDINAL WELD SHELL A, LONG WELD
B2.12	1-PZR-20	LONGITUDINAL WELD SHELL J, LONG WELD
B2.11	2-PZR-CIRC LHA	LOWER HEAD TO SHELL A
B2.11	2-PZR-CIRC DUH	SHELL D TO UPPER HEAD
B2.12	2-PZR-LONG A	LONGITUDINAL WELD SHELL A
B2.12	2-PZR-LONG D	LONGITUDINAL WELD SHELL D

For additional details on the licensee's request, please refer to the documents located at the ADAMS Accession Nos. identified above.

STAFF EVALUATION

1.0 Licensee's Basis for Proposed Alternative

The licensee referred to the results of the probabilistic fracture mechanics (PFM) analyses in the following Electric Power Research Institute (EPRI) report as the primary basis for proposing to increase the ISI interval for the requested components from 10 years to 30 years: non-proprietary EPRI report 3002015905, "Technical Bases for Inspection Requirements for PWR [Pressurized-Water Reactor] Pressurizer Head, Shell-to-Head, and Nozzle-to-Vessel Welds," December 2019 (ADAMS Accession No. ML21021A271). This report will be referred to as "EPRI report 15905" from this point forward.

The NRC staff's review focused on evaluating the PFM analyses in Section 8.3 of EPRI report 15905 and verifying whether the deterministic fracture mechanics (DFM) analyses in the report support the PFM results. The NRC staff reviewed the proposed alternative request for Salem Unit Nos. 1 and 2 as a plant-specific alternative. The NRC did not review EPRI

report 15905 for generic use, and this alternative request does not extend beyond the Salem Unit Nos. 1 and 2 plant-specific authorization.

2.0 Degradation Mechanisms

In Section 6.0 of Attachment 1 to the submittal, the licensee referred to the evaluation of potential degradation mechanisms in EPRI report 15905 and concluded that other than corrosion fatigue (also referred to as environmental assisted fatigue in the report) and mechanical/thermal fatigue, there were no active degradation mechanisms identified that significantly affect the long-term structural integrity of the PZR welds.

The NRC staff noted that the crack growth mechanism resulting from mechanical/thermal fatigue is fatigue crack growth (FCG), and that the effects of corrosion fatigue on FCG are included in the FCG rate selected for analyses (see Section 7 of this safety evaluation (SE)). The NRC staff finds the conclusion that corrosion fatigue and mechanical/thermal fatigue (both of which contribute to FCG) are the only active degradation mechanisms to be acceptable for the Salem Unit Nos. 1 and 2 plant-specific alternative request because: (1) FCG is known to be the dominant crack driving force in ferritic materials such as the PZR shell-to-head welds of Salem Unit Nos. 1 and 2 (see Section 5.1 of this SE); and (2) ferritic materials are known to be highly resistant to stress corrosion cracking under the operating conditions of the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

3.0 Overall PFM Approach

The PFM analyses in EPRI report 15905 were performed with the **PR**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction (PROMISE) Version 2.0 software. See Section 3.1 of this SE for a discussion of the verification and validation (V&V) of the software. The software will be referred to as PROMISE from this point forward unless otherwise noted.

The overall PFM approach in EPRI report 15905 is based on a Monte Carlo sampling technique in which PROMISE samples parameters with statistical distributions, also called random parameters, many times to calculate a probability. Each sampling of parameters is known as a trial or a realization (see Section 9.4 of this SE for a discussion of the number of realizations used in the analysis). For each realization, PROMISE performs a DFM analysis based on linear elastic fracture mechanics (LEFM), to calculate a time to failure to develop a histogram of failure times, which is, briefly stated, a tally of failure times. Section 8.3.2.9 of EPRI report 15905 defines failure as either rupture or leakage. Rupture is considered to occur when the applied stress intensity factor (SIF) exceeds plane strain crack initiation fracture toughness (K_{IC}). Leakage is considered to occur when the crack depth exceeds 80 percent of the wall thickness. From the histogram of failure times, PROMISE estimates the probability of failure (PoF) at a given time as the fraction of the total number of realizations that the computed failure time is less than the given time. The PoF is then determined on a per year basis and compared to an acceptance criterion of 1E-06 per year.

The NRC staff finds the overall PFM approach acceptable for the plant-specific alternative request for Salem Unit Nos. 1 and 2 because the Monte Carlo technique is a widely used and accepted technique for calculating probabilities, and counting times to failure is counting the number of failures (i.e., the probability that the failure time is less than a given time is equivalent to the probability that a failure would occur within that given time).

The NRC staff noted that the acceptance criterion of 1E-06 failures per year is tied to that used by the NRC staff in the development of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events." In that rule, the reactor vessel through-wall crack frequency (TWCF) of 1E-06 per year for a pressurized thermal shock event is an acceptable criterion because reactor vessel TWCF is conservatively assumed to be equivalent to an increase in core damage frequency, and as such meets the criteria in Regulatory Guide (RG) 1.174, "An Approach to for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." This assumption is conservative because a through-wall crack in the reactor vessel does not necessarily increase core damage. The discussion of TWCF is explained in detail in the technical basis document for 10 CFR 50.61a, NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," August 2007 (ADAMS Accession No. ML072830074).

The NRC staff also noted that the TWCF criterion of 1E-06 per year was generated using a very conservative model for reactor vessel cracking. The NRC staff finds that the licensee's use of 1E-06 failures per year based on the reactor vessel TWCF criterion is acceptable for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2 because the impact of a PZR vessel failure is less than the impact of a reactor vessel failure on overall risk. The NRC staff further noted that comparing the probability of leakage to the same criterion is conservative because leakage is less severe than rupture.

Lastly, the NRC staff noted that acceptance criterion of 1E-06 failures per year is lower, and thus more conservative, than the criterion the NRC staff accepted in proprietary report BWRVIP-05 "BWR [Boiling-Water Reactor] Vessel and Internals Project: BWR Reactor Pressure Vessel Weld Inspection Recommendation, September 1995"; non-proprietary report BWRVIP-108NP-A, "BWR Vessel and Internals Project: Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, October 2018" (ADAMS Accession No. ML19297F806); and non-proprietary report BWRVIP-241NP-A, "BWR Vessel and Internals Project: Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, October 2018" (ADAMS Accession No. ML19297G738). These EPRI reports were developed prior to or around the time the rules for PTS were reevaluated, and as such the acceptance criterion for failure frequency in the reports is based on the guidelines for PTS analysis in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors" that were available at the time. RG 1.154 was later withdrawn in 2011.

Based on the discussion above, the NRC staff finds the use of the acceptance criterion of 1E-06 failures per year for PoF acceptable for the Salem Unit Nos. 1 and 2 plant-specific alternative request.

3.1 *Software V&V*

In Section 8.0 of Attachment 1 to the submittal, the licensee stated that the alternative request for Salem Unit Nos. 1 and 2 uses PROMISE Version 2.0. The licensee also stated that the previous version of PROMISE, PROMISE Version 1.0, was used in another EPRI report referenced as the technical basis for an alternative request by Southern Nuclear Operating Company (SNC) (ADAMS Accession No. ML20253A311). As part of the review of SNC's alternative request, the NRC staff conducted an audit of PROMISE Version 1.0 in 2020 (at the time only this version of the software was available) to verify that it properly implemented PFM

principles and has undergone adequate V&V. During the audit, the NRC staff reviewed the V&V plan and the documents for the test cases that were performed to implement the plan. The NRC staff issued the audit summary report by letter dated December 10, 2020 (ADAMS Accession No. ML20258A002). The NRC staff issued its SE of the SNC submittal by letter dated January 11, 2021 (ADAMS Accession No. ML20352A155).

As documented in the audit summary report, the NRC staff requested benchmarking runs with another PFM software, VIPERNOZ, contained in Structural Integrity Associates (SIA) report 1900064.407.R2 (Enclosure 3 in ADAMS Accession No. ML20253A311). Even though the NRC staff has not formally accepted VIPERNOZ, it is the PFM software used in the BWRVIP-108 report for which the NRC staff has issued an SE dated December 19, 2007 (ADAMS Accession No. ML073600374). While SIA report 1900064.407.R2 was submitted as part of the plant-specific submittal by SNC, the benchmarking runs were performed with generic stresses instead of plant-specific stresses. The NRC staff reviewed the benchmark runs in 1900064.407.R2 and determined that the results showed adequate agreement between PROMISE Version 1.0 and VIPERNOZ for both probability of leakage values and probability of rupture values for different ISI scenarios. EPRI performed benchmarking runs with PROMISE Version 2.0 in Section 8.3.3.2 of EPRI report 15905. The NRC staff noted that with the benchmark of PROMISE Version 1.0, benchmarking of PROMISE Version 2.0 that is additional to the one performed in Section 8.3.3.2 of EPRI report 15905 is not necessary because of the adequate V&V performed for the difference between the two versions, as discussed next.

Because the NRC staff has already reviewed the V&V of PROMISE Version 1.0 as discussed above, the NRC staff determined that for the current alternative request for Salem Unit Nos. 1 and 2, only the difference between PROMISE Version 2.0 and PROMISE Version 1.0 needs to be reviewed. In Section 8.0 of Attachment 1 to the submittal, the licensee summarized the difference between PROMISE Version 2.0 and PROMISE Version 1.0: “The main difference between the two versions is that in PROMISE Version 1.0, the user-specified examination coverage is applied to all inspections, whereas in PROMISE Version 2.0, examination coverage can be specified by the user uniquely for each inspection.” In the supplement dated April 12, 2021, the licensee described the V&V performed for this difference. The NRC staff determined that the V&V of PROMISE Version 2.0 is adequate because the licensee demonstrated that the code change was properly implemented for only those cases where examination coverages for each inspection were specified.

Based on the above, the NRC staff finds for the Salem Unit Nos. 1 and 2 plant-specific alternative request that PROMISE Version 2.0 received adequate V&V, and therefore, is acceptable for use in the licensee’s plant-specific alternative request for the PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

4.0 Parameters Most Significant to PFM Results

In a PFM analysis, examples of the various input parameters that contribute to the final PoF value include crack dimensions, fracture toughness, stress, crack growth rate, and ISI schedule, all of which may be further defined by sub-parameters (such as the exponent term in the crack growth rate). Analysts typically use two sensitivity tools to understand the effects of the input parameters. Sensitivity analyses (SA) help identify the major contributors to the final PoF value, and sensitivity studies (SS) help in determining the impact of each parameter to the final PoF value.

In Section 8.3.4.2 of EPRI report 15905, EPRI performed SA to determine the dominant parameters that contribute to the probability of leak and rupture in certain welds of a representative PZR vessel. The results of these SA are in Tables 8-13 and 8-14 of the report for one of the locations analyzed. For probability of leakage, EPRI determined that the most dominant contributor is FCG rate coefficient, and for probability of rupture, EPRI determined that the most dominant contributor is fracture toughness.

The NRC staff reviewed the overall results of the SA in EPRI report 15905 with respect to the Salem Unit Nos. 1 and 2 plant-specific alternative request. Leakage is driven by growth of the postulated crack by the FCG rate, which is a measure of how fast the postulated crack would grow to 80 percent of the wall thickness; and FCG rate is proportional to the FCG rate coefficient. Thus, the NRC staff finds that the FCG rate coefficient being the dominant contributor to probability of leakage to be reasonable. Rupture is driven by applied SIF (which is driven by stress) or fracture toughness since applied SIF and fracture toughness (represented by K_{IC}) are the two main parameters in the governing expression in LEFM: applied SIF < K_{IC}. Thus, the NRC staff finds that fracture toughness being the dominant contributor to probability of rupture to be reasonable. The NRC staff noted that even though applied SIF did not come out as the dominant contributor in the SA in EPRI report 15905, it is one of the significant parameters reflected in the parameter of stress in the SS in EPRI report 15905, as discussed in the next paragraph.

In Section 8.3.4.3 of EPRI report 15905, EPRI performed SS on the following parameters: stress, fracture toughness, initial crack depth, number of flaws, flaw density, crack size distribution, FCG rate, probability of detection (POD), ISI schedule, and number of realizations. EPRI concluded that the most significant parameters are FCG rate, stress, and fracture toughness. As with the SA results for probability of leakage, the NRC staff finds the overall result of the SS on FCG rate reasonable for the Salem Unit Nos. 1 and 2 plant-specific alternative request since FCG rate is a measure of how fast the postulated crack would grow to 80 percent of the wall thickness. Similarly, as with the SA results for probability of rupture, the NRC staff finds the overall result of the SS on stress and fracture toughness reasonable for the Salem Unit Nos. 1 and 2 plant-specific alternative request since these are the parameters that directly affect the governing expression in LEFM.

During the audit of PROMISE Version 1.0, the NRC staff observed that ISI schedule and examination coverage have a significant impact on the PoF. The NRC staff requested two SIA letter reports that cover these topics, 1900064.406.R0 and 1900064.407.R2, which were included as Enclosures 2 and 3, respectively, in the SNC submittal (ADAMS Accession No. ML20253A311). Even though these two SIA letter reports were part of SNC's plant-specific alternative request, the impact of ISI schedule and examination coverage was a generic observation of the NRC staff on the PFM methodology.

The SA, SS, and the NRC staff's observations on the PROMISE software thus identified the following significant parameters or aspects of the PFM analyses that warrant a close evaluation: stress analysis, fracture toughness, FCG rate coefficient (or simply FCG rate), and effect of ISI schedule and examination coverage. The NRC staff discussed and closely evaluated each in the next four sections of this SE. The NRC staff also evaluated other parameters or aspects of the analyses in Section 9 of this SE.

5.0 Stress Analysis

5.1 *Selection of Components and Materials*

In Appendix A of Attachment 1 to the submittal, the licensee evaluated the plant-specific applicability of the components and materials selected and analyzed in EPRI report 15905 to the PZR shell-to-head welds of Salem Unit Nos. 1 and 2. The licensee showed that Salem Unit Nos. 1 and 2 met the component configuration and material criteria. The acceptability of meeting the criteria, however, depends on the acceptability of the component and material selection described in EPRI report 15905, which the NRC staff evaluated below.

In Sections 4.3 and 4.5.1 of EPRI report 15905, EPRI discussed the variation among PZR designs and selection of the shell-to-head and vessel head welds of a representative PZR vessel. EPRI used this selection for finite element analyses (FEA, see Section 5.4 of this SE) to determine stresses in the PZR shell-to-head welds, which the licensee referenced for the corresponding PZR shell-to-head welds requested for Salem Unit Nos. 1 and 2. In selecting the components, EPRI considered geometry, operating characteristics, materials, field experience with respect to service-induced cracking, and the availability and quality of component-specific information.

EPRI concluded that variations in the design of PZR vessels are not significant, and that the most important parameter, ratio of radius-to-thickness (R/t) can be addressed by sensitivity studies in the PFM evaluation. Table 4-2 of EPRI report 15905 shows the various R/t ratios considered.

The NRC staff reviewed Sections 4.3 and 4.5.1 of EPRI report 15905, and finds the PZR configurations selected in the report for stress analysis acceptable representatives for the corresponding PZR shell-to-head welds requested for the Salem Unit Nos. 1 and 2 plant-specific alternative request because differences in R/t ratios are small, and therefore, differences in stresses would be reasonably addressed through the SS on stress in EPRI report 15905. To verify the dominance of the R/t ratio, the NRC staff reviewed the through-wall stress distributions in Sections 7.1 and 7.2 of EPRI report 15905 to confirm that the pressure stress is dominant, which would confirm the dominance of the R/t ratio. The NRC staff finds that EPRI's conclusion about the R/t ratio being the dominant parameter in evaluating the various configurations to be acceptable for the Salem Unit Nos. 1 and 2 plant-specific alternative request since the pressure stress is the dominant stress as evidenced in Figures 7-10, 7-11, and 7-22 through 7-25 of EPRI report 15905.

In Section 5.1 of EPRI report 15905, EPRI discussed the material properties for the ferritic materials, SA-533, Grade B, Class 1, and SA-508, Class 2; and SA-240 Type 304 stainless steel assumed for all cladding material that were selected for the stress analysis. The NRC staff finds these materials acceptable for the Salem Unit Nos. 1 and 2 plant-specific alternative request because they are sufficiently similar to materials used in the Salem PZR vessel base metal and cladding.

Based on the review of Appendix A of Attachment 1 to the submittal, the NRC staff finds that Salem Unit Nos. 1 and 2 met the component configuration and material criteria in EPRI report 15905; therefore, the component configuration and materials of the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2 are acceptable.

5.2 *Selection of Transients*

In Appendices C and D of Attachment 1 to the submittal, the licensee evaluated the plant-specific applicability of the transients selected in EPRI report 15905 to the PZR shell-to-head welds of Salem Unit Nos. 1 and 2. The licensee stated that the Salem Unit Nos. 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those in Tables 5-6 and 5-10 of EPRI report 15905. The acceptability of Salem's transients meeting the transient criteria, however, depends on the acceptability of the transient selection described in EPRI report 15905, which the NRC staff evaluated below.

In Section 5.2 of EPRI report 15905, EPRI discussed the thermal and pressure transients under normal and upset conditions considered relevant to PZR shell-to-head welds. EPRI developed a list of transients for analysis, shown in Table 5-6 of EPRI report 15905, that is applicable to all PZR shell-to-head welds analyzed in the report, based on transients that have the largest temperature and pressure variations. EPRI stated that additional cycles of the loss-of-load transient addressed the transients not explicitly selected for analysis in EPRI report 15905. EPRI also developed a list of insurge/outsurge transients, shown in Table 5-9 of EPRI report 15905, that is applicable to the welds in the PZR bottom head, in addition to the general transients in Table 5-6 of EPRI report 15905. Insurge/outsurge transients are events that occur due to changes in the inventory of reactor coolant within the PZR resulting from the PZR's control of pressure of the reactor coolant system; these changes in reactor coolant inventory cause reactor coolant to flow in and out of the surge nozzle at the bottom of the PZR vessel.

The NRC staff reviewed the discussion of transients in Section 5.2 of EPRI report 15905, and determined that the transients defined in Tables 5-6 and 5-9 of EPRI report 15905 selected for analysis are reasonable for the Salem Unit Nos. 1 and 2 plant-specific alternative request because the transient selection was focused on those events with large temperature and pressure variations, thus conducive to FCG that is expected to occur in PWRs, including a set of insurge/outsurge transients applicable to the welds in the PZR bottom head to account for reactor coolant inventory changes within the PZR.

EPRI did not consider test conditions beyond a system leakage test. EPRI stated that since any pressure tests will be performed at operating pressure, no separate test conditions need to be included in the transient selection. The NRC staff reviewed the Salem Updated Final Safety Analysis Report and noted that Sections 5.2.1.5.10 and 5.2.1.5.11 (ADAMS Accession No. ML19360A116) specify up to 10 cycles of hydrostatic tests and 50 cycles of leak tests for a 40-year design life. In the supplement dated April 12, 2021, the licensee stated that it has performed operating system leakage tests instead of hydrostatic tests on the Salem Unit Nos. 1 and 2 PZRs following repair and replacement activities, and that leakage tests are expected to be performed for any potential future repairs on the Salem Unit Nos. 1 and 2 PZRs. The licensee explained that hydrostatic tests are performed during construction prior to initial plant startup, and stated that leakage tests are conducted as an integral part of the heatup process after refueling outages. The NRC staff finds for the Salem Unit Nos. 1 and 2 plant-specific alternative request that not including hydrostatic tests for the referenced analysis in EPRI report 15905 is acceptable because the licensee performs leakage tests instead. Also, the NRC staff determined for the Salem Unit Nos. 1 and 2 plant-specific alternative that since the leakage tests are integral to the heatup process, the number of cycles of leakage test need not be separate from the 102 and 107 projected 60-year cycles of the heatup/cooldown transient for Salem Unit Nos. 1 and 2, respectively, provided in Tables C-1 and C-2 in Attachment 1 to the submittal, and that these cycles are bounded by the 300 cycles of the heatup/cooldown transient assumed in Table 5-6 of EPRI report 15905.

The NRC staff needed confirmation that at the maximum pressures during test conditions at Salem Unit Nos. 1 and 2, the temperature of the PZR shell-to-head welds is high enough such that the upper shelf K_{IC} of value of 200 ksi√in assumed in EPRI report 15905 for fracture toughness is appropriate, considering the value of the nil-ductility reference temperature (RT_{NDT}) of 60 degrees Fahrenheit (°F) assumed in calculating K_{IC} in EPRI report 15905. The licensee's comparisons of applied SIF history with K_{IC} discussed in Section 6 of this SE resolved the NRC staff's note of low fracture toughness that can occur during the beginning and ending portions of the heatup/cooldown transient.

Based on the discussion above, the NRC staff finds that the leakage test conditions at Salem Unit Nos. 1 and 2 are adequately captured in the heatup/cooldown transient analyzed in EPRI report 15905.

EPRI did not evaluate faulted or emergency conditions separately. In Section 5.2 of EPRI report 15905, EPRI stated that the insurge/outsurge transients, along with the general transients, will bound other events, and that therefore, a faulted event is not specified for the PZR bottom head. The NRC staff determined this to be a reasonable conclusion because one of the insurge/outsurge transients includes a large change in temperature (330 °F) that would accommodate the stress state expected during a faulted event.

Based on the discussion above and the review of Appendices C and D of Attachment 1 to the submittal, the NRC staff finds that Salem Unit Nos. 1 and 2 met the transient criteria in EPRI report 15905; therefore, the transient loads for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2 are acceptable.

5.3 *Other Operating Loads*

In Section 8.2.2.3.2 of EPRI report 15905, EPRI discussed the cosine distribution assumed for the through-wall residual stress due to welding. EPRI stated that this residual stress distribution has been used in past projects, particularly BWRVIP-108 for which the NRC staff has accepted and issued an SE dated December 19, 2007 (ADAMS Accession No. ML073600374).

BWRVIP-108 is the PFM-based technical basis for the reduction of the number of nozzles inspected in BWR pressure vessels on which some of the inputs for EPRI report 15905 were based. The NRC staff noted that the residual stress distribution in BWRVIP-108 is for welding in thick-walled vessels. The NRC staff finds the cosine distribution EPRI assumed for the through-wall residual stress due to welding acceptable for the shell-to-head welds of the PZR vessel of Salem Unit Nos. 1 and 2 because they are in the thick-walled locations of the PZR vessel. Finally, the NRC staff noted that treatment of the through-wall residual stress as a constant (non-random) parameter in EPRI report 15905 is reasonable because it only has a mean load effect on FCG and does not affect the range of applied load that is the main driver of FCG.

The NRC noted that the PZR shell-to-head welds of Salem Unit Nos. 1 and 2 are cladded, and that therefore, the effect of clad residual stress needed to be included. The licensee included the effect of clad residual stress using Equation 8-1 of EPRI report 3002015905. The NRC staff determined that using Equation 8-1 of EPRI report 15905 would adequately account for clad residual stress because, as can be seen in the equation, the clad residual stress decreases with increasing temperature, which is the expected behavior of clad residual stress with respect to temperature. However, the NRC staff observed that adding the clad residual stress determined from Equation 8-1 of EPRI report 15905 with the FEA stresses described in Section 7 of EPRI

report 15905 for the cladded welds could result in a lower net stress within the cladding because the effect of differential thermal expansion between the cladding and base metal (an effect that results in a compressive stress within the cladding) is included twice: first in the thermal stress from the FEA (see Section 5.4 of this SE) and second in Equation 8-1 of EPRI report 15905, which includes the differential thermal expansion effect in addition to residual stress due to the welding of the cladding.

To resolve the doubling of the thermal expansion differential effect, the NRC staff further looked into how the clad residual stress affects the postulated flaws in the PZR shell-to-head welds analyzed in EPRI report 15905. The depth of the postulated flaws is described by the distribution derived from flaw data from Pressure Vessel Research User's Facility (PVRUF) project (see Section 9.1 of this SE). This distribution is shown in Equation 8-2 of EPRI report 15905. Based on this postulated flaw distribution, the depths of flaws that are evaluated in the PFM analysis 90 percent of the time are 0.0787 inch or less. The thickness of the cladding in the modeled PZR vessel in EPRI report 15905 is 0.125 inch in the upper PZR shell-to-head welds and 0.063 inch in the lower PZR shell-to-head welds.

For the lower PZR shell-to-head welds, most of the postulated flaw depth is greater than the clad thickness of 0.063 inch, which means that the crack tip is in the ferritic base metal that has a lower fracture toughness compared to the stainless steel cladding, and that therefore, accounting for the effect of differential thermal expansion between the cladding and base metal only once could have an impact on the final probability of rupture values. The NRC staff calculated the total applied SIF for a 0.1-inch deep flaw due to thermal stress, pressure stress, and clad residual stress at a temperature of 70 °F when the value of Equation 8-1 of EPRI report 15905 is maximum. The resulting total applied SIF is 55 ksi$\sqrt{\text{in}}$. The NRC staff noted that including the pressure stress is conservative since the pressure is low when temperature is low. The NRC staff determined that for the lower PZR shell-to-head welds even if the effect of differential thermal expansion between the cladding and base metal was accounted for only once, it would have little impact on the final probability of rupture values since the total applied SIF is less than the 80 ksi$\sqrt{\text{in}}$ performed in one of the SS on toughness in EPRI report 15905.

For the upper PZR shell-to-head welds, since 90 percent of time the postulated flaw depth is 0.0787 inch or less, which is well within the cladding thickness, the postulated flaw would be within the stainless steel cladding, where the fracture toughness is much higher than that of the ferritic steel base metal, making it very unlikely that the total applied stress would lead to failure if the flaw stays within the cladding after growth. If the flaw propagates into the ferritic base metal, similar to the discussion in the previous paragraph, the total applied SIF would be less than the 80 ksi$\sqrt{\text{in}}$ performed in one of the SS on toughness in EPRI report 15905. Accordingly, the NRC staff determined that even if the effect of differential thermal expansion between the cladding and base metal was accounted for only once, there would be little or no impact on the final probability of rupture values.

Based on the discussion above, the NRC staff finds the treatment of other loads described in this section of the SE acceptable for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

5.4 *Finite Element Analyses*

In Section 7 of EPRI report 15905, EPRI discussed the FEA to determine stresses due to internal pressure and thermal transients for the selected geometries discussed in Section 5.1 of

this SE. The NRC staff reviewed the modeling details (elements used, boundary conditions, symmetry assumptions, etc.) and finds that they are consistent with standard FEA practice.

The NRC staff also reviewed the stress contour plots and the through-thickness stress distribution and finds them acceptable for the plant-specific alternative request. For instance, the NRC staff verified the hoop stress due to a unit pressure shown in the bottom plots of Figures 7-10 and 7-11 of EPRI report 15905.

The NRC staff noted that the through-wall stress distribution plots for the thermal transients that have temperature drops analyzed in EPRI report 15905 show compressive stresses at the inner surface (see Figure 7-10 of EPRI report 15905, for example). Tensile stresses at the inside surface are typically expected for transients that have temperature drops, such as the insurge/outsurge transients. In the supplement dated April 12, 2021, the licensee explained that for the transients with temperature drops, the PZR starts hot and that the differential thermal expansion between the stainless steel cladding and the low alloy steel base metal causes significant compression on the inside surface at the start of these transients. The licensee included figures that showed that the inside surface stress became less compressive during the transient and explained the temperature drop was not large enough for the stresses to become tensile except for the cooldown portion of the heatup/cooldown transient. The NRC staff verified independently that the differential thermal expansion between the stainless cladding and low alloy steel base metal can generate a compressive stress on the inside surface at hot conditions and that the compressive stress can remain during the transient if the temperature drop is not large enough. Accordingly, the NRC staff determined that having compressive stresses on the inside surface for the transients in question is reasonable, even though the licensee's stress value on the inside surface is more compressive than the NRC staff's value. The NRC staff noted that the compressive stress in the FEA caused by the differential thermal expansion between the stainless cladding and low alloy steel base metal is due to the effect of two different adjacent materials and does not account for the residual stresses within the clad generated from the welding process of the clad. An iterative procedure in the FEA would need to be performed in order to determine a stress-free temperature that would adequately simulate the effect of clad residual stress in the FEA. The NRC staff discussed the topic of clad residual stress in Section 5.3 of this SE.

Based on the discussion above, the NRC staff determined that the pressure and thermal stresses calculated through FEA in EPRI report 15905 are acceptable for referencing for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

6.0 <u>Fracture Toughness</u>

In Sections 8.2.2.6 and 8.3.2.7 of EPRI report 15905, EPRI assumed for fracture toughness of ferritic materials an upper shelf K_{IC} value of 200 ksi$\sqrt{in}$ based on the upper-shelf fracture toughness value in the ASME Code, Section XI, A-4200. EPRI treated K_{IC} as a random parameter normal distribution with a mean value of 200 ksi$\sqrt{in}$ and a standard deviation of 5 ksi$\sqrt{in}$, stating that these assumptions are consistent with the BWRVIP-108 project. As discussed in Section 5.1 of this SE, Salem Unit Nos. 1 and 2 meets the material criteria in EPRI report 15905, and thus the NRC staff determined that the fracture toughness parameters above are applicable to Salem Unit Nos. 1 and 2.

The NRC staff had accepted the treatment of upper shelf K_{IC} as a random parameter in BWRVIP-108 through sensitivity studies on K_{IC} that the NRC staff requested, which changed the standard deviation to less than 5 ksi$\sqrt{in}$ and greater than 5 ksi$\sqrt{in}$ (see December 19, 2007, SE

of BWRVIP-108; EPRI report 15905 also includes SS on K_{IC} as discussed below). Apart from this, EPRI report 15905 states that the RT_{NDT} assumed for the PZR welds is 60 °F, which the NRC staff noted would have some conservative effect in the assumption of the upper-shelf K_{IC} value. For these reasons, the NRC staff finds that the mean and standard deviation values of upper shelf K_{IC} used in EPRI report 15905 are reasonable for the Salem Unit Nos. 1 and 2 plant-specific alternative request, even though statistical distributions that would more accurately account for the uncertainty in upper-shelf fracture toughness should have been used.

EPRI further explained that an upper shelf K_{IC} value of 200 ksi√in can be used for fracture toughness since the minimum temperature from all applicable transients is sufficiently high (i.e., the temperature in the PZR welds analyzed in EPRI report 15905 stays in the range in which the upper-shelf value of 200 ksi√in is applicable).

The NRC staff noted that the temperature of the heatup/cooldown transient in Table 5-6 of EPRI report 15905 selected for analysis in the report can be as low as 70 °F. The NRC staff further noted that therefore a fracture toughness value lower than the upper-shelf fracture toughness of 200 ksi√in assumed in the analysis may exist during the beginning and ending portions of the heatup/cooldown transient. In the supplement dated April 12, 2021, the licensee showed applied SIF history plots for the Heatup/Cooldown transient for the limiting case, PRSHC-BW-2C, in EPRI report 15905. The plots showed that the applied SIF did not exceed the lowest value of K_{IC} of about 100 ksi√in, which the licensee determined from the plant-specific RT_{NDT} value 10 °F for the Salem Unit Nos. 1 and 2 PZRs. The staff noted that the applied SIF history was for deep and long flaws, and the occurrence of deep and long flaws with the flaw distribution used in the analysis (see Section 9.1 of this SE) is rare during the Monte Carlo sampling. The licensee's comparison of applied SIF history with K_{IC} resolves the NRC staff's note of low fracture toughness during the beginning and ending portions of the heatup/cooldown transient.

Based on the discussion above and the discussion in Sections 5.1 and 5.2 of this SE which confirmed that the materials and transient loads are acceptable for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2, the NRC staff finds the fracture toughness model in EPRI report 15905 acceptable for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

7.0 FCG Rate

In Sections 8.3.2.6 of EPRI report 15905, EPRI stated that the FCG rate for ferritic steels, as defined in the 2017 Edition of the ASME Code, Section XI, Appendix A, paragraph A-4300, is used in the evaluation. The NRC staff verified that the 2017 Edition of the ASME Code, Section XI, is the latest edition incorporated by reference in 10 CFR 50.55a. The NRC staff also confirmed that the FCG rate in A-4300 in the 2017 Edition of ASME Code, Section XI, is the same as that in the 2004 and 2013 Editions (no addenda) of ASME Code, Section XI, because the 2004 and 2013 Editions of ASME Code, Section XI, are the codes of record for fourth and fifth ISI intervals of Salem Unit Nos. 1 and 2. The NRC staff noted that the FCG rate in ASME Code, Section XI, A-4300 is applicable to both BWR and PWR.

The FCG rate is defined with a log-normal distribution with the median value defined as the FCG rate in ASME Code, Section XI, A-4300, and with a value of 0.467 for the uncertainty parameter. The NRC staff finds the uncertainty parameter of 0.467 acceptable for the Salem Unit Nos. 1 and 2 plant-specific alternative request because it is based on over 1,000 FCG rate data for low alloy steels.

In Section 8.3.4.1 of EPRI report 15905, EPRI stated that assuming the A-4300 curve as the median curve is conservative since the actual data from which the A-4300 curve is based on represent the 95 percent confidence limit of the data. The NRC staff clarifies that 95 percent confidence limit here means that the A-4300 curve bounds the median of the data 95 percent of the time; it does not mean that the A-4300 curve is the 95th percentile of the data. The NRC staff determined that because of the amount of available data for ferritic FCG rate, however, the difference between the 50 percent confidence limit on the median and the 95 percent confidence limit on the median would likely be small. Thus, the NRC staff determined for the Salem Unit Nos. 1 and 2 plant-specific alternative request that assuming the A-4300 curve as the median curve would only be slightly conservative.

EPRI stated that the associated threshold on the FCG rate is also log-normally distributed and that the log-normal distributions on the rate and threshold are consistent with the approach used in the xLPR, a PFM software sponsored by the NRC and EPRI. The NRC staff confirmed that the FCG rate in xLPR received adequate V&V. The NRC staff noted that the FCG rate is the rate defined in A-4300 with a statistical distribution around it since the FCG rate is treated as a random parameter. In Section 8.3.4.3.7 of EPRI report 15905, EPRI performed a SS on the effect of FCG rate on probability of leakage by replacing the A-4300 FCG rate with the FCG rate used in BWRVIP-108. The result of the study showed that the A-4300 FCG rate led to a much higher probability of leakage. The NRC staff noted that the FCG rate in ASME Code, Section XI, A-4300 is applicable to both BWR and PWR and therefore, is applicable to Salem.

Based on the discussion above, the NRC staff finds that the A-4300 FCG rate used in the analyses is acceptable for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

8.0 ISI Schedule and Examination Coverage

In Section 6.0 of Attachment 1 to the submittal, the licensee stated that for Salem Unit Nos. 1 and 2, preservice inspection (PSI) examination followed by four 10-year ISI examinations (PSI + 10 + 20 + 30 + 40) have been performed and included the inspection history and examination coverage results for the requested PZR shell-to-head welds. The NRC staff noted that for two of these welds, 2-PZR-CIRC LHA and 2-PZR-CIRC DUH, the fourth 10-year ISI examination has not yet been performed and is scheduled to be performed. The licensee is proposing an alternative that would extend the ISI interval to 30 years after the first four 10-year ISI examinations (i.e., an alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70).

In Section 8.3.4.1.2 of EPRI report 15905, EPRI discussed the effect of various ISI schedules on the PoF results, and in Section 8.3.5 of the report, discussed inspection (i.e., examination) coverage.

The NRC staff noted the impact of ISI schedule on the PoF values, as shown in Table 8-12 of EPRI report 15905. In the audit summary report for PROMISE (see Section 3.1 of this SE), the NRC staff observed how ISI is implemented in the software, as described in the following: the number and frequency of ISI are input into the software; at the specified times of ISI, flaws are either detected or not detected with the chance of detection/non-detection given by the POD curve (see Section 9.2 of this SE for further discussion of the POD curve). If detected, a flaw is assumed to be repaired or properly dispositioned, and thus, cannot cause failure; if not detected, the flaw continues to grow, and thus, can lead to failure. The NRC staff determined this to be a better approach than applying an adjustment factor to the failure probabilities since the effect of the POD curve would be propagated into the failure probabilities each time ISI is implemented. As discussed in Section 3.1 of this SE, the NRC staff requested additional

benchmarking runs with VIPERNOZ contained in SIA report 1900064.407.R2 (Enclosure 3 in ADAMS Accession No. ML20253A311). These benchmarking runs were performed with different ISI schedules and generic stresses. The comparison plots in Figures 1 through 4 of SIA report No. 1900064.407.R2 showed adequate agreement between PROMISE Version 1.0 and VIPERNOZ. Implementation of ISI did not change in PROMISE Version 2.0 (see Section 3.1 of this SE). Thus, because of the adequate implementation and benchmarking of ISI, the NRC finds that the PoF values in EPRI report 15905 adequately included the effect of ISI schedule. Since the licensee referenced the PFM results in EPRI report 15905 for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2, the NRC staff finds that the licensee adequately included the effect of ISI schedule on the PoF values for the PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

In Section 8.3.5 of EPRI report 15905, EPRI stated that it assumed 100 percent inspection of the required volume (i.e., 100 percent examination coverage) of the PZR welds analyzed in the reports during each of the ISI scenarios evaluated in the reports, which assumes 100 percent examination coverage during PSI. EPRI further explained that based on its statements on the PSI-only examinations in Section 8.3.4.1 of EPRI report 15905, by performing examinations with 100 percent coverage during PSI, no other examinations are needed for safe plant operation for 80 years. EPRI further stated that based on this, any additional ISI examinations after PSI would reduce the already low PoF values, and that, therefore, the PFM evaluations with 100 percent examination coverage assumed for all ISI also apply to partial (i.e., less than 100 percent, examination coverage). In Section 10 of this SE, the NRC staff explained its non-acceptance of the licensee's and EPRI's conclusion on PSI-only examinations. Thus, the NRC staff determined that partial examination coverage plays a vital role in the final PoF values.

In the audit summary report for PROMISE, the NRC staff observed how the software implements examination coverage for a case with 50 percent coverage. In a given PFM evaluation, the POD curve is not applied for approximately 50 percent of the number of realizations at the specified times of ISI, and thus for 50 percent of the realizations, a postulated flaw would continue to grow. The NRC staff determined that this an acceptable approach for implementing examination coverage since its effect would be propagated into the failure probabilities each time an ISI is implemented.

In Section 6.0 of Attachment 1 to the submittal, the licensee discussed the examination history of the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2, which shows that the examination coverage for weld 2-PZR-CIRC DUH was as low as 37.2 percent. In Section 8.3.5 of EPRI report 15905, EPRI included PFM results that show the effect of 50 percent examination coverage on PoF values. The resolution of the 37.2 percent examination coverage for weld 2-PZR-CIRC DUH is discussed in Section 10 of this SE.

Therefore, because examination coverage was adequately implemented and PFM results for less than 100 percent examination coverage were included in the PoF calculations as discussed above, the NRC finds that the licensee adequately addressed the effect of examination coverage on the PoF values for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

9.0 Other Considerations

9.1 *Initial Flaw Depth and Length Distribution*

In Section 8.3.2.2 of EPRI report 15905, EPRI stated that the flaw distribution derived from flaw

data from the PVRUF vessel was applied to the PZR vessel in the analyses. NUREG-6471, "Characterization of Flaws in U.S. Reactor Pressure Vessels," November 1998 (ADAMS Accession No. ML112510316), states that the PVRUF vessel is from an unused PWR vessel and that the PVRUF data are from fabrication flaws in the PVRUF vessel weldment. The NRC staff noted that the PVRUF depth distribution represented by Equation 8-2 of EPRI report 15905 consisted of mostly small flaws that are inner surface-breaking.

Figure 4-4 of EPRI report 15905 shows the PZR vessel lower head model used in the analyses referenced for the Salem Unit Nos. 1 and 2 shell-to-head welds in the PZR lower head; the figure shows that the PZR vessel cylindrical shell (vertical portion) is relatively thick, but the PZR lower head is only 2.55 inches thick. The NRC staff noted that the nominal thickness of PWR vessels is 8 inches, and that since the PVRUF flaw data are based on a PWR vessel, the data may not be appropriate for vessels much thinner than 8 inches, such as the PZR lower head, since welding thinner vessels is different from welding thick vessels. Furthermore, Table A-1 in Attachment 1 to the submittal states that the upper and lower head of the Salem Unit Nos. 1 and 2 PZRs are made of SA-216, Grade WCC, which is different from SA-533 and SA-508, Class 2, materials typically used for reactor pressure vessels. Section 8.3.2.2 of EPRI report 15905 included only a general discussion of the applicability of PVRUF to a PZR vessel, stating that "even though the PVRUF data were based on a reactor pressure vessel, they can be applied to a pressurizer vessel because both are large-diameter vessels fabricated from similar plate and forging process and from the same materials (SA-533 and SA-508 Class 2)." The NRC staff needed additional information to determine if the PVRUF flaw data is appropriate for the Salem Unit Nos. 1 and 2 PZR vessel lower shell-to-head welds, given that the modeled PZR lower heads are much thinner than reactor pressure vessels, and given the difference in materials of the Salem Unit Nos. 1 and 2 PZR vessel lower heads from the materials of reactor pressure vessels.

In the supplement dated April 12, 2021, the licensee assessed the PVRUF flaw data for applicability to the Salem Unit Nos. 1 and 2 PZR shell-to-head welds in the lower head. In this assessment, the licensee considered five other initial flaw distributions in addition to the PVRUF distribution in Section 8.3.2.2 of EPRI report 15905. The licensee stated that collectively, the six initial flaw distributions represent distributions that were developed for vessel and piping that consider the relevant geometrical parameter (i.e., thickness), and different materials and manufacturing processes for such components. The licensee explained that two of the six distributions were included for SS to address the component-specific effect of geometry (i.e., thickness), and unknowns related to differences in materials and manufacturing processes. The first of the two shifted the PVRUF distribution in a manner that makes the sampling of initial flaw depth conservative relative to PVRUF, and the second of the two assumed a constant initial flaw depth of 25 percent of the thickness. The licensee performed PFM analyses using all six flaw distributions for the limiting case, PRSHC-BW-2C, in EPRI report 15905 and demonstrated that the resulting probabilities of rupture and leakage values were insensitive to initial flaw depth. The NRC staff reviewed the licensee's assessment of the applicability of PVRUF to the Salem Unit Nos. 1 and 2 PZR vessel lower shell-to-head welds, and finds it acceptable because (1) the licensee appropriately considered the effects due to differences in geometry, materials, and manufacturing processes with conservative initial flaw depth distributions, and (2) the licensee demonstrated that the resulting probabilities of rupture and leakage values were insensitive to initial flaw depth. Thus, the NRC staff determined for the Salem Unit Nos. 1 and 2 plant-specific alternative request that applying the PVRUF initial flaw depth distribution to the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2 is acceptable.

In Section 8.3.2.2 of EPRI report 15905, EPRI also described the length distribution used in the

PFM analyses. EPRI cited NUREG/CR-6817, "A Generalized Procedure for Generating Flaw-Related Inputs for the FAVOR Code," March 2004 (ADAMS Accession No. ML040830499), for the log-normal distribution for the flaw length. As the NRC staff observed in the audit summary report for PROMISE, the flaw data for the length distribution was derived from the most conservative of three sets of flaw data, and as such, the NRC staff finds for the Salem Unit Nos. 1 and 2 plant-specific alternative request that the length distribution is acceptable for the analysis results referenced for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

9.2 *Probability of Detection*

In Section 8.3.2.3 of EPRI report 15905, EPRI stated that the POD curve used in the analyses was the same POD curve used in the BWRVIP-108 analyses. The NRC staff confirmed that the POD curve in Figure 8-6 of EPRI report 15905 is the same as the POD curve in BWRVIP-108. The NRC staff noted that the welds and nozzles analyzed in BWRVIP-108 were associated with the reactor pressure vessel and that the POD curve was, therefore, developed based on the ultrasonic testing (UT) requirements in the ASME Code, Section XI, Appendix VIII (this is also reflected in the discussion of POD in the December 19, 2007, SE of BWRVIP-108).

The PZR welds analyzed in EPRI report 15905 are associated with the PZR vessel for which the UT requirements of ASME Code, Section V, apply. The NRC staff noted that, in practice, the POD curve based on the UT requirements of the ASME Code, Section V, could be lower than the POD curve based on the UT requirements of the ASME Code, Section XI, Appendix VIII. To evaluate the acceptability of the Appendix VIII-based POD curve on the PZR vessel for which the UT examination requirements of ASME Code, Section V apply, the NRC staff assessed the PVRUF cumulative probability distribution shown in Equation 8-2 of EPRI report 15905 against the Appendix VIII-based POD curve in Figure 8-6 of EPRI report 15905. The PVRUF distribution represented by Equation 8-2 of EPRI report 15905, in effect, says that there is about a 90 percent probability that the initial flaw depth used in the PFM analyses is equal to or less than 0.0787 inches. This flaw depth is on the lower portion (left side) of the Appendix VIII-based POD curve. While the NRC staff expects that, in practice, a POD curve based on the ASME Code, Section V, could be lower than the Appendix VIII-based POD curve, it would not be much lower for flaw depths equal to or less than 0.0787 inches, which are flaw depths that are analyzed 90 percent of the time and for which the POD is already very low at about 18 percent.

Based on the discussion above and that POD is not one of the parameters that significantly affects the PFM results, the NRC staff determined for the Salem Unit Nos. 1 and 2 alternative request that a Section V-based POD curve would have minimal impact on the PFM results compared to an Appendix VIII-based POD curve, and that therefore, the Appendix VIII-based POD curve is adequate for use in the PFM analyses referenced for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

9.3 *Models*

In Section 8.2.2.5 of EPRI report 15905, EPRI described the fracture mechanics models used in the analyses. For both semi-elliptical circumferential and axial surface cracks in a cylindrical configuration, EPRI employed SIF models that are similar to the crack models used to analyze postulated flaws in the nozzle-to-shell welds in BWRVIP-108 and BWRVIP-241, for which the NRC staff has approved and issued SEs.

For the postulated flaws in the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2, the NRC staff compared the applied SIF value for a postulated axial flaw in a cylinder to the applied SIF value for the same flaw in a sphere. Even though the PZR shell-to-head welds are at the juncture of the cylindrical and spherical (i.e., head) portion of the PZR shell, the comparison shows the effect of the curvature of the head on applied SIF. The NRC staff determined that the applied SIF for the cylindrical model is slightly higher than the applied SIF for the spherical model. Therefore, the NRC staff finds for the Salem Unit Nos. 1 and 2 plant-specific alternative request that the cylindrical SIF models in the EPRI report 15905 are appropriate for the postulated flaws in the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

9.4 *Uncertainty*

In Section 8.3.1.2 of EPRI report 15905, EPRI considered both aleatory uncertainty (random or inherent uncertainty) and epistemic uncertainty (uncertainty due to state of knowledge) and stated that these uncertainties entailed two sampling loops: an aleatory loop and an epistemic loop. In Section 8.3.4.1 of EPRI report 15905, EPRI stated that it considered all random parameters aleatory because they are conservative or based on large sets of data (for example, the FCG distribution was developed from over 1,000 fatigue datapoints in PWR water environments). EPRI performed 10 million aleatory realizations and 1 epistemic realization for the PFM analyses.

The NRC staff noted that representing all variables as aleatory will result in probabilities that represent the mean of the distribution. From the distribution of results presented, the NRC staff noted that the 50th percentile (median) was very close to the results when the licensee assumed all aleatory realizations (mean). This is expected for a distribution that is normally distributed but may be different for a skewed distribution.

In the audit summary report for PROMISE, the NRC staff documented observations on percent error and implementation of aleatory and epistemic realizations. With regard to the observation on percent error, the NRC staff notes that large percent errors that result from probabilistic analyses where only one failure happens in 10 million realizations can be impactful if the results approach the acceptance criteria. Assuring sufficient realizations and proper sampling of the input space will reduce the error with these calculations. In addition, the overuse of conservative inputs in a probabilistic analysis can mask the importance of other random variables and should be avoided. However, since the limiting location for the base case in EPRI report 15905 had probabilities of leakage and rupture more than two orders of magnitude below the acceptance criterion of 1E-06 per year, the NRC staff finds for the Salem Unit Nos. 1 and 2 alternative request that the large uncertainty in the low probability results reflected by the large percent error is acceptable.

In Table 8-9 of EPRI report 15905, EPRI indicated that there were no uncertainties in the transient stresses. The NRC staff finds for the Salem Unit Nos. 1 and 2 alternative request that

treating transient stresses as constant rather than random is acceptable since the transients were selected based on large temperature and pressure variations, as discussed in Section 5.2 of this SE.

Based on the discussion above, the NRC staff finds for the Salem Unit Nos. 1 and 2 alternative request that the licensee's handling of uncertainty is acceptable for the analysis results referenced for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

9.5 *Convergence*

In Section 8.3.4.3.10 of EPRI report 15905, EPRI conducted an SS to determine if the number of realizations resulted in a converged solution in the PoF values. The results in Table 8-31 of EPRI report 15905 indicates little difference in PoF values between 10^7 and 10^8 realizations. Based on these results, the NRC staff finds for the Salem Unit Nos. 1 and 2 plant-specific alternative request that the number of realizations used in the analyses, 10^7 realizations, is acceptable for the results referenced for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2. This number of realizations is acceptable even though the uncertainty is high for those cases where only one failure occurs within an analysis, as described in Section 9.4 of this SE.

9.6 *Flaw Density*

In Section 8.3.2.2 and Table 8-9 of EPRI report 15905, EPRI indicated that 1.0 flaw per weld is used in the PZR analyzed. EPRI stated that these values are consistent with those the NRC staff approved in BWRVIP-108. The NRC staff determined in the December 19, 2007, SE of BWRVIP-108 that based on a surface-breaking flaw density of 0.01 flaw per cubic foot (flaw/ft^3), 1.0 flaw per weld is conservative for the nozzle weld configurations analyzed in BWRVIP-108. Similarly, using a surface-breaking flaw density of 0.01 flaw/ft^3 per weld and the volumes of the subject PZR welds of Salem Unit Nos. 1 and 2 estimated from the PZR shell diameter in Table A-2 of Attachment 1 to the submittal, the NRC staff determined that 1.0 flaw per weld is adequate for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

9.7 *DFM Analysis*

In Section 8.2 of EPRI report 15905, EPRI performed DFM analyses with an initial flaw depth based on the maximum depth specified in the ASME Code, Section XI, acceptance standards and average values of all other parameters considered random in the PFM analyses. All analyzed locations resulted in many years to reach leakage (80 percent of the component thickness), the least being 433 years. No locations reached an applied SIF greater than an allowable SIF value of 100 ksi√in (mean fracture toughness in the PFM evaluations, 200 ksi√in, with a safety factor of 2). Thus, the NRC staff determined that overall, the DFM analyses support the PFM analyses, and that therefore, along with the NRC staff's determinations in the preceding sections on the PFM aspects of the EPRI analyses as they apply to Salem, the PFM analyses in EPRI report 15905 can be referenced for the Salem Unit Nos. 1 and 2 plant-specific alternative request.

10.0 PFM Results Relevant to Alternative SC-I4R-20 0

In Section 6.0 of Attachment 1 to the submittal, the licensee stated that based on the PFM results, after PSI, no other inspections are required for up to 60 years of plant operation to meet the acceptance criterion of 1E-06 failures per year. A similar observation is in Section 8.3.4.1.1

of EPRI report 15905, which states that performing only PSI examination without any other post-PSI examinations is acceptable for 80 years of plant operation while maintaining plant safety. The NRC staff does not find either general conclusion acceptable since it does not account for the effect of the combination of the most significant parameters or the added uncertainty of low probability events.

The NRC staff determined that since the PFM analyses in EPRI report 15905 were based on representative PZR vessels, the uncertainties on the different parameters (which are different from the sampling uncertainty discussed in Section 9.4 of this SE) should be taken into account, especially those from the significant parameters of stress and fracture toughness before a general conclusion can be made on PSI-only examinations. Also, the effects of other significant parameters such as examination coverage should be considered. As an example, even for a case more favorable than PSI-only examination, such as PSI + 10 + 20 + 40 + 60, the probability of rupture at 80 years for the limiting location changed from 1.25E-09 per year to 3.18E-07 per year (Table 8-32 of EPRI report 15905). While the NRC staff acknowledged that this study assumed conservative values for stress and fracture toughness simultaneously (thereby accounting for uncertainties in these two parameters), the NRC staff also noted that had the same study been performed for the PSI-only case, the probability of rupture values would have been much higher, and that only one of the two parameters could easily lead to probability of ruptures greater than 1E-06 per year.

Given the discussion on uncertainty above and in Section 9.4 of this SE, the NRC staff determined that uncertainty in the PFM results need to be addressed through sufficient realizations and proper sampling before general conclusions can be considered for the PSI-only cases. Lastly, the NRC staff observed that PSI-only examinations, as compared to the proposed alternative of PSI + 10 + 20 + 30 + 40 + 70, would have a much more adverse effect on risk-informed principles, particularly since PSI-only examinations would remove any future condition monitoring needed for risk-informed decision making.

As discussed in Section 8 of this SE, the licensee is seeking the alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70. Therefore, the NRC staff determined that PFM results for PSI + 10 + 20 + 30 + 40 + 70 are the results relevant to the licensee's proposed alternative.

The NRC staff noted that even though EPRI 15905 report does not have PoF results for PSI + 10 + 20 + 30 + 40 + 70, it has results for PSI + 20 + 40 + 60 or PSI + 10 + 20 + 40 + 60, either of which would bound the former since ISI is implemented more times in the former. Therefore, the NRC staff evaluated the PFM results in the SS in Section 8.3.4.3 of EPRI report 15905 relevant to the proposed alternative ISI schedule of PSI + 10 + 20 + 30 + 40 + 70 by assessing the results for PSI + 20 + 40 + 60 or PSI + 10 + 20 + 40 + 60.

Table 8-32 of EPRI report 15905 shows the probability of rupture results for the SS on the combined effect of fracture toughness and stress. These probability of rupture results are for an ISI schedule of PSI + 10 + 20 + 40 + 60, which bound the licensee's proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the reasons the NRC staff previously stated. As shown in Table 8-32 of EPRI report 15905, the limiting probability of rupture is 3.18E-07 per year, which is below the criterion of 1E-06 per year. The NRC staff noted that that if fracture toughness was set to the base case values of 200 ksi√in with standard deviation of 5 ksi√in, which the NRC staff found acceptable in Section 6 of this SE, the limiting case would have much more margin from the criterion of 1E-06 per year. This larger margin is shown in the SS on stress in Table 8-17 of EPRI report 15905, which shows that even with a stress multiplier of 1.80, the limiting probability of rupture is 2.50E-09 per year.

The results in Tables 8-17 and 8-32 of EPRI report 15905 discussed above assume 100 percent examination coverage. As mentioned in Section 8 of this SE, the licensee showed in Section 6.0 of Attachment 1 to the submittal that the examination coverage for weld 2-PZR-CIRC DUH of the Salem Unit 2 PZR was as low as 37.2 percent. The licensee performed an additional sensitivity run specifically with a 37.2 percent examination coverage, but only reported the probability of leakage value. Similarly, Table 8-33 of EPRI report 15905 shows the effect of examination coverage, but only probability of leakage values were reported. In the supplement dated April 12, 2021, the licensee stated that for the limiting case, PRSHC-BW-2C, in Table 8-33 of EPRI Report 15905, the probability of rupture value is 1.25E-09 per year after 80 years with a 50 percent examination coverage and an inspection schedule of PSI + 20 + 40 + 60. The NRC staff determined that this result adequately addresses the 37.2 percent examination coverage for weld 2-PZR-CIRC DUH because the result was much lower than the criterion of 1E-06 failures per year and because there was no change in probability of rupture values going from 100 percent to 50 percent examination coverage. As such, the result provides reasonable assurance that, for the plant-specific alternative request for Salem Unit Nos. 1 and 2, the limiting case in EPRI report 15905 referenced for weld 2-PZR-CIRC DUH is not sensitive to examination coverage.

Finally, the NRC staff noted since the licensee's proposed alternative is through 60 years of operation, the probability values should be based on 60 years of operation. The PFM results in EPRI report 15905 discussed above are for 80 years of operation, and at 60 years of operation the results could be up to 80/60 = 1.3 times larger since the number of failures would be divided by 60 years instead of 80 years (assuming the number of failures have been reached by 60 years). As discussed in Section 3 of this SE, PoF at a given time is estimated as the fraction of the total number of realizations that the computed failure time is less than the given time. In short, this means that PoF is the number of failure times within a given time divided by the total number of realizations. For instance, if the given time is 60 years, PoF is the number of failure times that are less than 60 years divided by the total number of realizations. Since the number of failure times could be reached before 60 years, the PoF value could be the same at 60 years and at 80 years. Since the licensee's proposed alternative is through 60 years of operation, this PoF value should be divided by 60 years instead of 80 years to obtain the PoF per year value. The staff determined that this factor of 1.3 has no impact on the NRC staff's discussion of the PFM results in EPRI report 15905 in the preceding paragraphs. Thus, the NRC staff determined that the PFM analyses in EPRI report 15905 adequately address uncertainties in the PoF values relevant to the licensee's proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2.

Based on the discussion above, the staff finds that the proposed alternative of PSI + 10 + 20 + 30 + 40 + 70 for the requested PZR shell-to-head welds of Salem Unit Nos. 1 and 2 would result in a PoF per year that is below the acceptance criterion of 1E-06 per year.

CONCLUSION

The NRC staff has determined that the proposed alternative in the licensee's request referenced above would provide an acceptable level of quality and safety.

The NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

The NRC staff authorizes the use of proposed alternative SC-I4R-20 0 at Salem Generating Station Units 1 and 2 for the remainder of the fourth 10-year ISI interval and through the following fifth 10-year ISI interval, which is scheduled to end on December 31, 2030.

The NRC's authorization of the proposed alternative does not infer or imply the approval of EPRI report 15905 for generic use.

All other ASME Code, Section XI, requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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Date: June 10, 2021

James G. Danna

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SUBJECT: SALEM GENERATING STATION UNIT NOS. 1 AND 2 – AUTHORIZATION AND SAFETY EVALUATION FOR ALTERNATIVE REQUEST NO. SC-I4R-20 0 (EPID L-2020-LLR-0103) DATED JUNE 10, 2021

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