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PSEG
Nuclear LLC

10 CFR 50.55a

April 12, 2021
LR-N21-0024

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Generating Station Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: Response to Request for Additional Information for Proposed Alternative for Examination of ASME Section XI, Examination Category B-B, Item Number B2.11 and B2.12

References:
1. PSEG Letter LR-N20-0041, "Proposed Alternative for Examination of ASME Section XI, Examination Category B-B, Item Number B2.11 and B2.12," dated August 5, 2020 (ADAMS Accession No. ML20218A587)
2. NRC E-mail, "Salem - Final RAI RE: Alternative for Examination of ASME Section XI, Category B-B, Item Number B2.11and B2.12 (L-2020-LRR-0103)," dated February 11, 2021 (ADAMS Accession No. ML21043A144)

In Reference 1, PSEG Nuclear LLC (PSEG) requested NRC approval of proposed relief request SC-I4R-200 for Salem Generating Station, Units 1 and 2 (Salem). The proposed relief is requesting an alternative to volumetric examination of Pressurizer circumferential and longitudinal shell-to-head welds to extend the inspection frequency to 30 years for Salem Units 1 and 2.

In Reference 2, the NRC staff provided PSEG with a Request for Additional Information (RAI) regarding the Reference 1 relief request. Attachment 1 to this submittal provides the responses to the RAI.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this matter, please contact Brian Thomas at 856-339-2022.

Sincerely,

Paul R. Duke, Jr.
Manager - Licensing

PSEG Nuclear LLC

Attachment 1: Response To Requests for Additional Information Salem Generating Station Units Nos. 1 and 2 Regarding Alternative for Examination of ASME Section XI, Category B-B, Item Number B2.11 And B2.12, EPID L-2020-LRR-0103

cc: Administrator, Region I, NRC
NRC Senior Resident Inspector, Salem
James Kim, Project Manager, NRC
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L. Marabella, Corporate Commitment Tracking Coordinator
T. Cachaza, Salem Commitment Tracking Coordinator

Response to Requests for Additional Information
Salem Generating Station Units Nos. 1 and 2
Regarding Alternative for Examination of ASME Section XI,
Category B-B, Item Number B2.11 and B2.12
EPID L-2020-LRR-0103

By letter dated August 5, 2020 (ADAMS Accession No. ML20218A587), PSEG Nuclear LLC (PSEG, the licensee), submitted to the United States Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for pressurized water reactor (PWR) pressurizer (PZR) circumferential and longitudinal shell-to-head welds (SHWs) of Salem Generating Station (Salem) Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.55a(z)(1), the licensee proposed to increase the ISI interval for the subject components to 30 years, from the current ASME Code Section, Section XI requirement of 10 years. 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. The licensee referred to the analyses in non-proprietary Electric Power Research Institute (EPRI) Report No. 3002015905, "Technical Bases for Inspection Requirements for PWR Pressurizer Head, Shell-to-Head, and Nozzle-to-Vessel Welds", December 2019 (ADAMS Accession No. ML21021A271) to support the proposed alternative in the submittal. The licensee also included an applicability evaluation of EPRI report 3002015905 to Salem Units 1 and 2 in the submittal. The NRC staff (the staff) needs to issue requests for additional information (RAIs) to complete its review of the licensee's proposed alternative.

RAI 1

Regulatory Basis

The NRC has established requirements in 10 CFR Part 50 to protect the structural integrity of structures and components in nuclear power plants. Among these requirements are the ISI requirements of Section XI of the ASME Code incorporated by reference in 10 CFR Part 50.55a to ensure that adequate structural integrity of the PZR vessel is maintained through the service life of the vessel. Therefore, the regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the subject PZR SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue

Table 5-6 of EPRI report 3002015905 shows that the temperature of the pressurizer (TPZR in the table) applicable to the Examination Item Numbers B2.11 and B2.12 PZR SHWs of Salem Units 1 and 2 requested in the submittal ranges from 70°F to 653°F for the Heatup/Cooldown transient. Section 8.2.2.6 of EPRI report 3002015905 explains that a fracture toughness (KIC) set at the upper shelf value of the ASME Code KIC curve, 200 ksi√in, may be used. In the audit summary report for PROMISE Version 1.0 (ADAMS Accession No. ML20258A002, Item No. 2.e.iii), the NRC staff observed that for the steam generator feedwater nozzle, KIC could be as low as 130 ksi√in during the Heatup/Cooldown transient and that this lower value of KIC is addressed by the sensitivity study on fracture toughness in Tables 8-13 and 8-14 of non-proprietary EPRI Report No. 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections", April 2019 (ADAMS Accession No. ML19347B107). The NRC staff noted that the lowest KIC value, 80 ksi√in, used in the sensitivity study on fracture toughness in EPRI report 3002015905 is the same as the lowest KIC value used in the sensitivity study on fracture toughness in EPRI report 3002014590. However, the NRC staff noted that the pressure for the B2.11 and B2.12 PZR SHWs of Salem Units 1 and 2 is more than twice the pressure analyzed for the SG feedwater nozzle in EPRI report 3002014590 since these welds are on the primary

side of the reactor coolant system. Since the pressure is more than twice than what was previously analyzed for the SG feedwater nozzle, the staff cannot determine how much lower than 200 ksi√in the KIC value could be during the ramp periods in the beginning and ending of Heatup/Cooldown transient for the subject PZR SHWs.

Request
Given the NRC staff's observations discussed above for the B2.11 and B2.12 PZR SHWs of Salem Units 1 and 2, explain how the sensitivity study on fracture toughness in EPRI report 3002015905 addresses the low KIC value that could exist during the ramp periods in the beginning and ending of the Heatup/Cooldown transient for the subject PZR SHWs.

PSEG Response:

From the deterministic fracture mechanics (DFM) results in Table 8-4 of EPRI Report 3002015905, the critical Case ID for Item Nos. B2.11 and B2.12 is PRSHC-BW-2C (both with and without surge nozzle piping moments). This Case ID also shows the most sensitivity to fracture toughness in the probabilistic fracture mechanics (PFM) evaluation as presented in Tables 8-15, 8-16 and 8-32 of EPRI Report 3002015905. Therefore, Case ID PRSHC-BW-2C was selected for an additional evaluation to respond to this RAI request.

The stress intensity factor (K) and temperature histories at the inside surface for Case ID PRSHC-BW-2C for the Heatup/Cooldown transient (including RCS pressure) are provided in Figure RAI-1-1 for three flaw depths that span the range of depths covered in the evaluation (a/t = 0.2, 0.5 and 0.8). Based on a review of the supporting calculations, the maximum aspect ratio (crack half-length over depth, or c/a) from the DFM evaluation performed in Section 8.2 of EPRI Report 3002015905 for this Case ID ranged from approximately 1.7 to 3.6. A conservative (longer flaw) aspect ratio, c/a, of 20 was selected for use in this evaluation.

The stress intensity factor time histories were calculated for the three flaw depths for the Heatup/Cooldown transient, and the fracture toughness history for the transient was calculated using the equation for K_{IC} in Paragraph A-4200 of Nonmandatory Appendix A of ASME Code, Section XI. A maximum plant-specific RT_{NDT} of 10°F was used for the Salem Units 1 and 2 pressurizers based on a review of all plant-specific pressurizer certified material test reports (CMTRs) and fabrication inspection reports. As shown in Figure RAI-1-1, the calculated applied Salem-specific stress intensity factors are bounded with margin by the corresponding K_{IC} calculated as a function of temperature throughout the transient, for all three flaw depths.

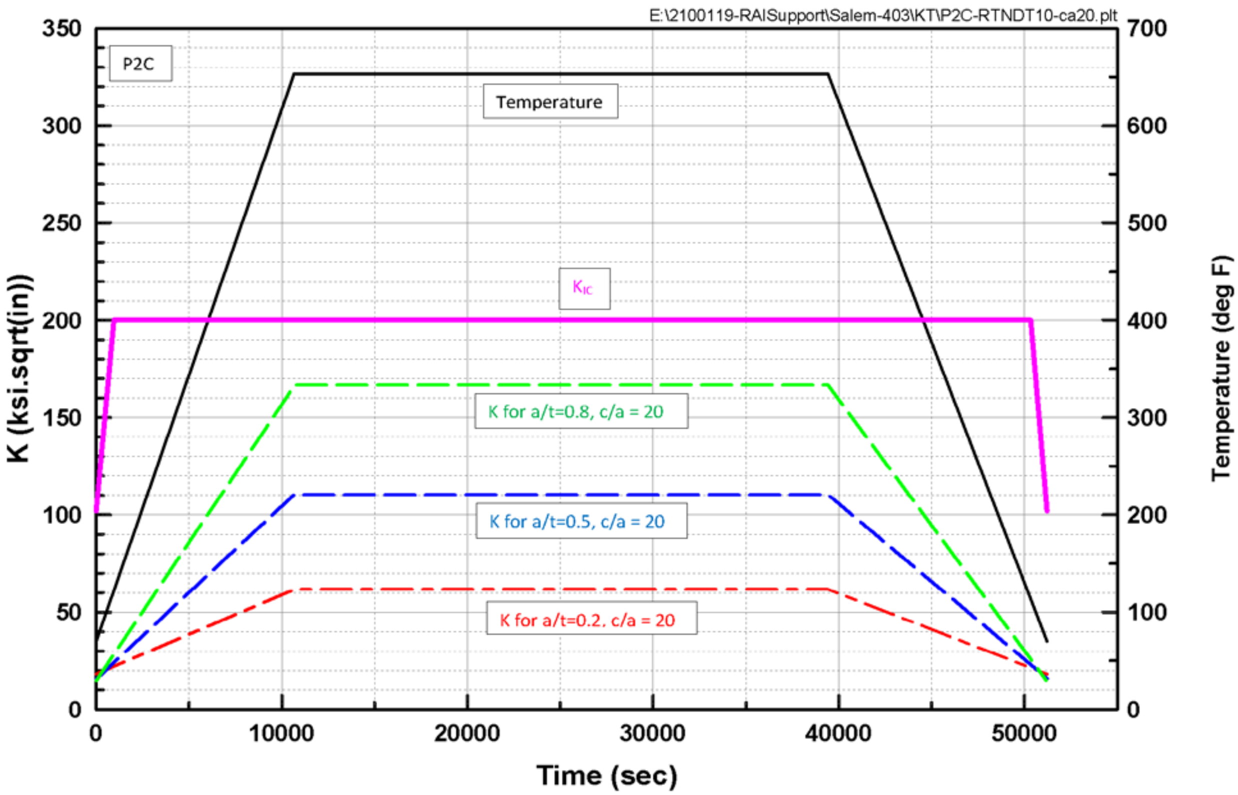

Figure RAI-1-1: Applied K vs. Fracture Toughness as a Function of Temperature for Case ID PRSHC-BW-2C
(Using Plant-Specific Maximum Pressurizer Material RT_{NDT} Value of 10°F)

RAI 2

Regulatory Basis

The regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the requested PZR SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue

In Section 5.2 of EPRI report 3002015905, EPRI stated that it did not consider test conditions beyond a system leakage test in the analyses, and stated that since any pressure tests will be performed at operating pressure, no separate test conditions need to be included in the analyses because the test conditions are captured in the other transients included in the analyses. Even though the pressure test conditions are not included in the analyses, the NRC staff determined that the appropriate temperature conditions for an upper shelf fracture toughness (KIC) value of 200 ksi√in assumed in the EPRI report should exist during the hydrostatic and leak tests at Salem Units 1 and 2.

Request

Confirm that at the maximum pressures during the hydrostatic and leak tests of Salem Units 1 and 2, the temperature of the subject PZR SHWs of Salem Units 1 and 2 is high enough such

that the upper shelf KIC of value of 200 ksi√in assumed in EPRI report 3002015905 for fracture toughness is appropriate, considering the value of the nil-ductility reference temperature (RTNDT) of 60°F assumed in calculating KIC in EPRI report 3002015905.

PSEG Response:

With respect to system tests, leakage tests in accordance with ASME Code, Section XI, Examination Category B-B are conducted after each refueling outage for Salem Units 1 and 2. The leakage tests are conducted while the plant is in Mode 3, as an integral part of the plant heatup process after refueling outages.

Hydrostatic tests refer to pressurization tests that are performed during construction prior to initial plant startup. Paragraph IWA-4540(a) of Section XI of the ASME Code requires an operating system leakage test or a system hydrostatic test following any major pressurizer repairs. PSEG has previously performed operating system leakage tests on the Salem Units 1 and 2 pressurizers following repair and replacement activities, rather than hydrostatic tests. PSEG would also expect to perform operating leakage testing following any potential Salem Units 1 and 2 pressurizer repairs in the future.

The foregoing activities are typical practice for many PWRs. Therefore, hydrostatic tests were not considered in EPRI Report 3002015905, and only leakage tests were evaluated as part of the Heatup/Cooldown transient.

The Heatup and Cooldown transients were evaluated together as one single transient, as noted in Section 7.1.2.3 of EPRI Report 3002015905. As indicated in Table 5-6 of EPRI Report 3002015905, 300 cycles of heatup/cooldown were assumed for 60 years of operation. Each of these events include pressurization from atmospheric pressure to operating pressure, and depressurization to atmospheric pressure. Because the leakage tests are conducted as an integral part of the plant heatup process, no additional cycles were included solely for leakage testing. Therefore, 300 cycles of heatup/cooldown, which include integral leakage tests, were considered in the EPRI report. The projected number of heatup/cooldown (with integral leakage tests) events for Salem Units 1 and 2 for 60 years of operation is 102 and 107 cycles, respectively, as identified in Tables C-1 and C-2 of Attachment 1 of Reference 1. Therefore, the projected 102 and 107 heatup/cooldown cycles (which also account for leak tests) for the Salem Units 1 and 2 pressurizers over 60 years of operation are bounded by the corresponding 300 cycles evaluated in EPRI Report 3002015905.

As discussed in the response to RAI 1, the applied stress intensity factors for the Salem Units 1 and 2 pressurizer welds throughout the Heatup/Cooldown transient (which includes leak tests) are bounded by the upper shelf fracture toughness (K_{IC}) value of 200 ksi√in assumed in EPRI Report 3002015905. The evaluation described in the response to RAI 1 considered the maximum plant-specific value for RT_{NDT} of 10°F to be appropriate for the Salem Units 1 and 2 pressurizers.

<u>RAI 3</u>

Regulatory Basis

The regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the requested PZR SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is referenced, and

thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue
The through-wall stress distribution plots for the thermal transients in Figures 7-10 and 7-11, 7-22 through 7-25 of EPRI report 3002015905 for the subject PZR welds show compressive stresses at the inner surface. However, tensile stresses at the inside surface are typically expected for transients that have temperature drops, such as the Insurge/Outsurge transients listed in Table 5-9 of EPRI report 3002015905 and the Loss of Load transient in Table 5-6 of EPRI report 3002015905.

Request
To assure that the thermal stress behavior in the subject PZR welds analyzed is reasonable, explain if the thermal stresses on the inside surface shown in the referenced figures of EPRI report 002015905 become tensile at times other than those shown in the figures.

PSEG Response:

The general trends are similar in all the through-wall stress profiles in Figures 7-10, 7-11, 7-22, 7-23, 7-24 and 7-25 of EPRI Report 3002015905. Therefore, the limiting stress path was selected for illustration in this response. Based on the results of the deterministic fracture mechanics (DFM) evaluation summarized in Section 8.2.3 and Table 8-4 of EPRI Report 3002015905, the limiting stress path is Path P2 for the pressurizer lower head-to-vessel weld. Path P2 is shown in Figure 7-9 of EPRI Report 3002015905. Inside surface stress histories were generated for limiting stress Path P2 for all six of the evaluated transients listed in Tables 5-6 and 5-9 of EPRI Report 3002015905 (Heatup/Cooldown, Loss of Load, and Insurge/Outsurge Groups 1 through 4 transients). These six plots are shown in Figures RAI-3-1 through RAI-3-6. The stress histories in these figures can be compared to the stress profiles shown for Path P2 in Figure 7-11 of EPRI Report 3002015905.

For the Heatup/Cooldown transient stress histories shown in Figure RAI-3-1, the inside surface stress is zero at the start of the transient because there is no differential thermal expansion between the stainless steel cladding and the low alloy steel base material for the assumed, stress-free temperature of 70°F. Then, a significant compressive stress develops during the heatup because of the differential thermal expansion between the cladding and the base material as the temperature increases. The compressive stress reaches a maximum steady state value after 10,000 seconds when the heatup is complete. The stress then reverses during the cooldown that begins just before 40,000 seconds and becomes tensile at the end of the cooldown near the end of the transient (at 50,000 seconds).

All other transients (the Loss of Load and the four Insurge/Outsurge Groups) start with the pressurizer hot and therefore, the inside surface at the start of these transients is in significant compression due to the differential thermal expansion between the stainless steel cladding and the low alloy steel base material based on the stress-free temperature of 70°F. As shown in Figures RAI-3-2 through RAI-3-6, the stresses on the inside surface during these transients become less compressive because of the cooldowns that occur in each of the transients, but the temperature drops are not high enough during the transients for the inside surface stresses to become tensile.

Therefore, the thermal stress behavior in the analyzed pressurizer welds are reasonable in that they follow expected behaviors based on the inputs used in the evaluation.

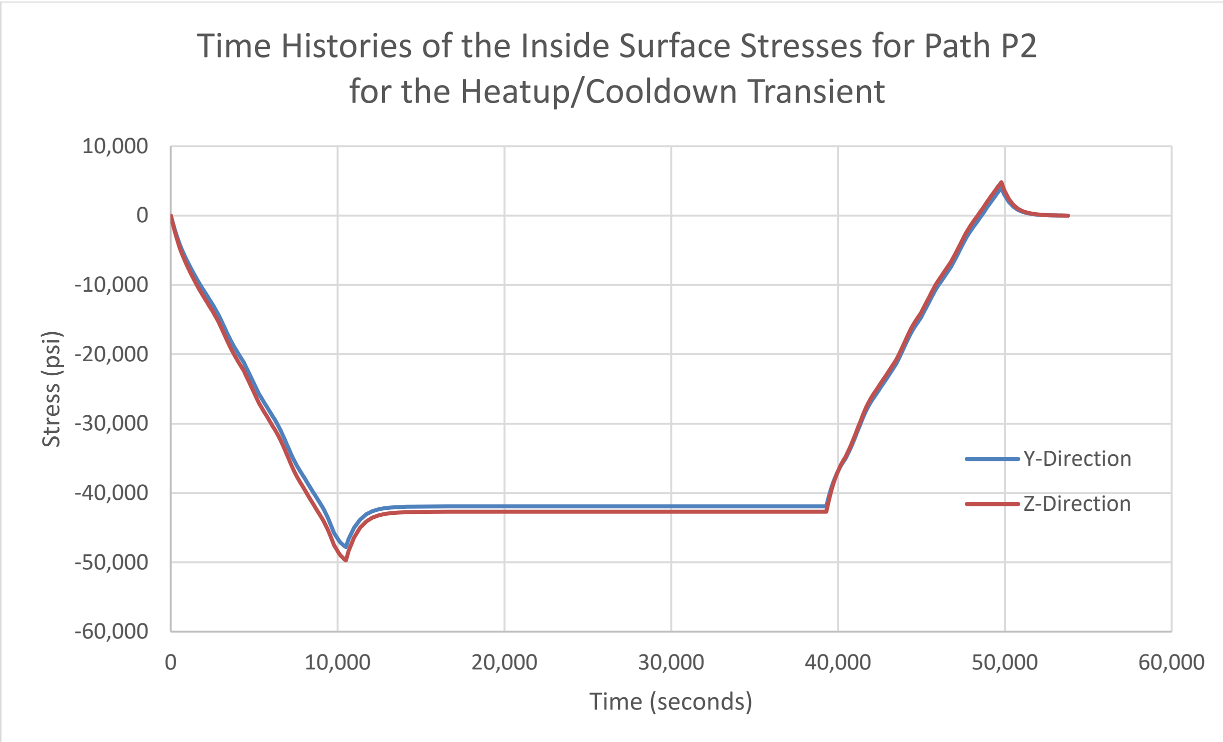

Figure RAI-3-1: Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2 for the Heatup/Cooldown Transient

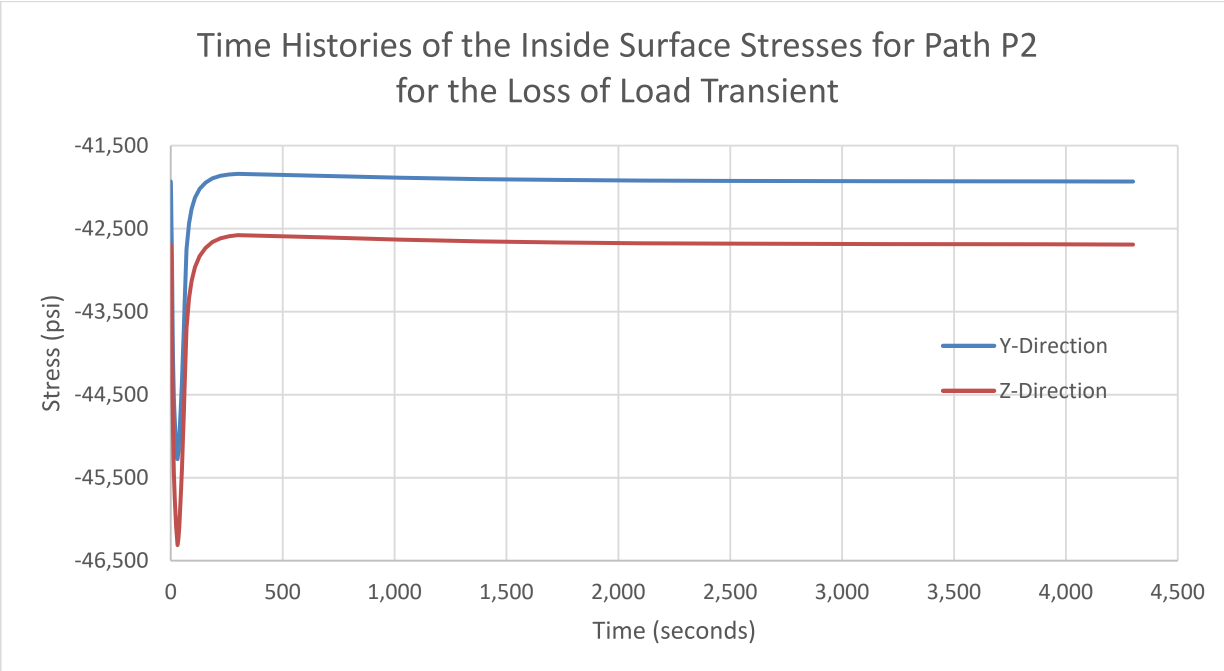

Figure RAI-3-2: Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2 for the Loss of Load Transient

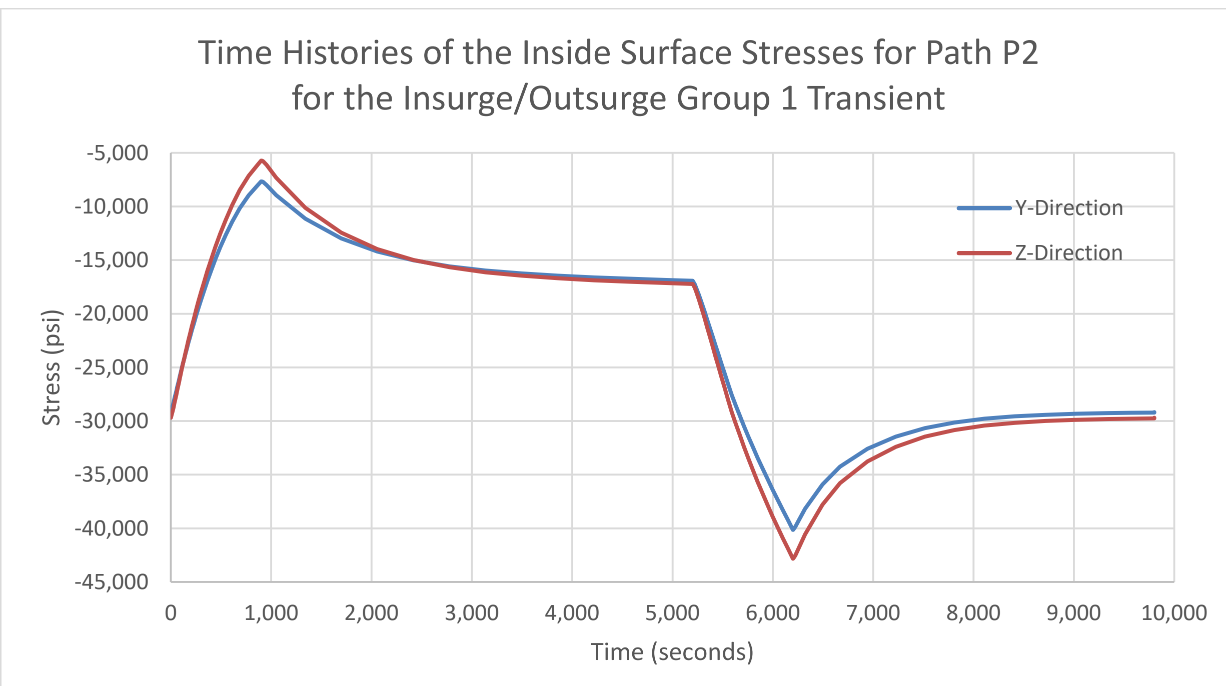

Figure RAI-3-3: Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2 for the Insurge/Outsurge Group 1 Transient

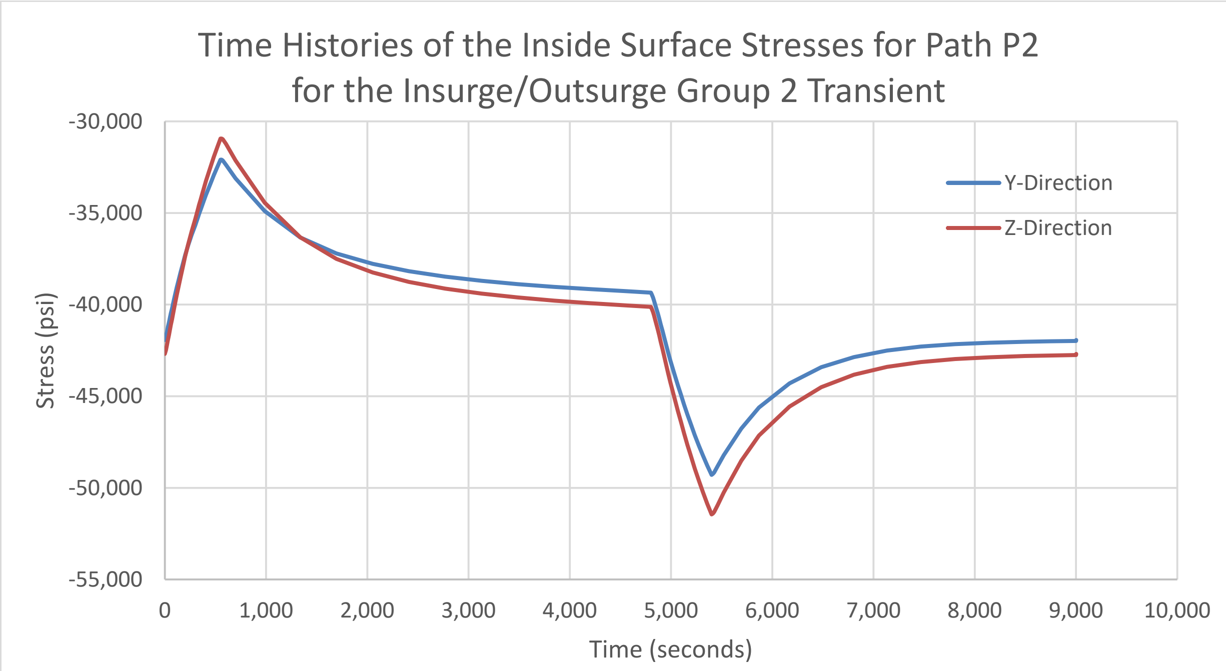

Figure RAI-3-4: Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2 for the Insurge/Outsurge Group 2 Transient

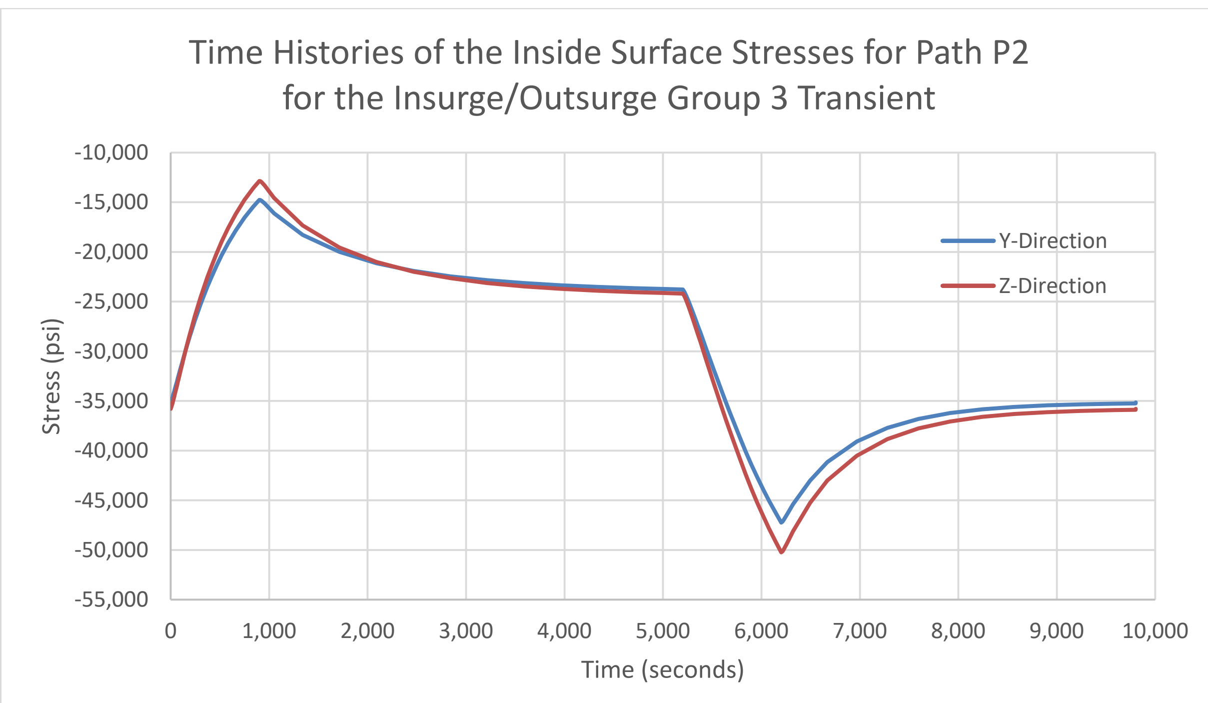

Figure RAI-3-5: Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2 for the Insurge/Outsurge Group 3 Transient

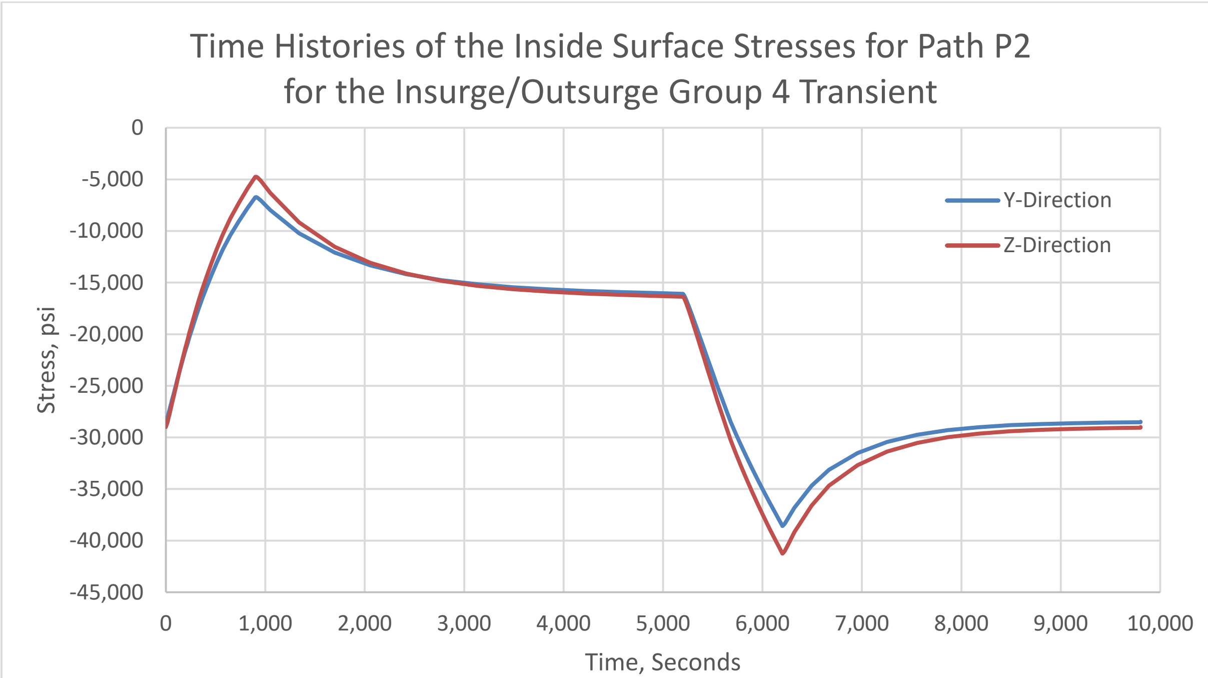

Figure RAI-3-6: Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2 for the Insurge/Outsurge Group 4 Transient

<u>RAI 4</u>

Regulatory Basis

The regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the requested PZR vessel lower SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue

In Section 8.3.2.2 of EPRI report 3002015905, EPRI stated that the flaw distribution derived from flaw data from the Pressure Vessel Research User's Facility (PVRUF) vessel was applied to the PZR vessel in the analyses. NUREG-6471 “Characterization of Flaws in U.S. Reactor Pressure Vessels,” (ADAMS Accession No. ML112510316, Reference 89 of EPRI report 3002015905) states that the PVRUF vessel is from an unused pressurized water reactor (PWR) vessel and that the PVRUF data are from fabrication flaws in the PVRUF vessel weldment. The nominal thickness of PWR vessels is 8 inches. Figure 4-4 of EPRI report 3002015905 shows the PZR vessel lower head model used in the analyses referenced for the Salem Units 1 and 2 SHWs 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 staff noted that the PVRUF flaw data may not be appropriate for vessels much thinner than 8 inches 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 Units 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

3002015905 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)." This discussion does not provide the staff sufficient information to determine that the PVRUF flaw data is appropriate for the Salem Units 1 and 2 PZR vessel lower SHWs, given that the modeled PZR lower heads are much thinner than reactor pressure vessels, and given the difference in materials of the Salem Units 1 and 2 PZR vessel lower heads from the materials of reactor pressure vessels.

Request
Justify that the flaw distribution based on the PVRUF flaw data from a reactor pressure vessel is appropriate for the Salem Units 1 and 2 PZR vessel lower SHWs.

PSEG Response:

To assess appropriateness of the PVRUF flaw data for application to the Salem Units 1 and 2 pressurizer lower head shell welds, various initial flaw distributions available in the literature were collected. These distributions were compared and evaluated for their potential application to the pressurizer. The flaw distributions that were collected are as follows:

1. PVRUF. This distribution is applicable to thick-walled, ferritic pressure vessel components, and is the same distribution used to evaluate the pressurizer welds as identified in Section 8.3.2.2 of EPRI Report 3002015905.
2. PVRUF shifted by a factor of 1.25. This distribution applied a conservative multiplication factor of 1.25 to the PVRUF distribution to account for reduced component thickness[(1)]. The intent of this shift was to account for the reduced pressurizer lower head wall thickness compared to a reactor pressure vessel (RPV).
3. Marshall. This distribution is applicable to thick-walled, ferritic pressure vessel components; it pre-dates the PVRUF distribution.
4. Chapman. This distribution is applicable to ferritic piping components. There are two distributions that reflect different welding processes.
    a. Manual metal arc welding (MMAW)
    b. Tungsten inert gas welding (TIG)
5. xLPR. This distribution is applicable to dissimilar metal piping butt welds between a ferritic nozzle and stainless steel piping using Alloy 82/182 weld metal.
6. Theoretical. This distribution assumes a conservatively uniform flaw distribution with a constant flaw depth of 25% of the wall thickness. The length is allowed to vary consistent with the description of the initial flaw distribution in Section 8.3.2.2 of EPRI Report 3002015905. In each realization, the crack depth is assumed to be 25% of wall thickness while the crack length is randomly selected.

These six flaw distributions collectively represent previously-developed flaw distributions (Nos. 1, 3, 4, and 5) for vessel and piping components that consider the relevant (different) geometry (thickness), materials and manufacturing processes for such components. In addition, theoretical distributions (#2 and #6) are included to perform sensitivity studies that address other geometries, materials and manufacturing unknowns that may be specific to other components relative to vessel components for which the PVRUF distribution applies. For

(1) The multiplication factor moved the X-axis (flaw depth, *a*) of the PVRUF flaw distribution to the right by a factor of 1.25 for the same *P(>a)* values, thus making the initial flaw depth conservative during the sampling.

pressurizers, the geometry, material and fabrication requirements are similar to those for RPVs, although the pressurizer wall thickness for the lower head region where the shell head welds reside is smaller compared to the cylindrical portion of an RPV. In addition, the Salem Units 1 and 2 pressurizer lower heads are carbon steel castings fabricated using SA-216, Grade WCC material rather than the rolled plate used to fabricate the PVRUF RPV from which the initial fabrication flaw distribution was developed and used in EPRI Report 3002015905.

Five of the above initial flaw size distributions are plotted in Figure RAI-4-1, where the probability of finding a flaw greater in size than depth, *a*, or *P(>a)*, is plotted as a function of the flaw depth, *a*. Flaw distribution #6 (the "Theoretical" flaw distribution) would plot as a vertical line in this figure. For example, for the pressurizer lower head, the vertical line would occur at a crack depth, *a*, of 0.64 inches for all *P(>a)* values (based on 25% of the minimum pressurizer lower head thickness of 2.55 inches shown in Table 4-4 of EPRI Report 3002015905).

It is noted that vessel flaw distributions (PVRUF or the shifted PVRUF) are considered to be most appropriate for application to the pressurizer because the pressurizer is a vessel with similar geometry, material and fabrication practices as those used to fabricate RPVs. On the other hand, the xLPR flaw distribution is applicable to dissimilar metal welds, which are not present in the subject pressurizer welds. Nevertheless, useful insights can be gained by comparing the probabilistic fracture mechanics (PFM) results from all six of the above flaw distributions to estimate the sensitivity in the PFM results from differences in the geometry, material, and fabrication practices that were assumed in each flaw distribution model.

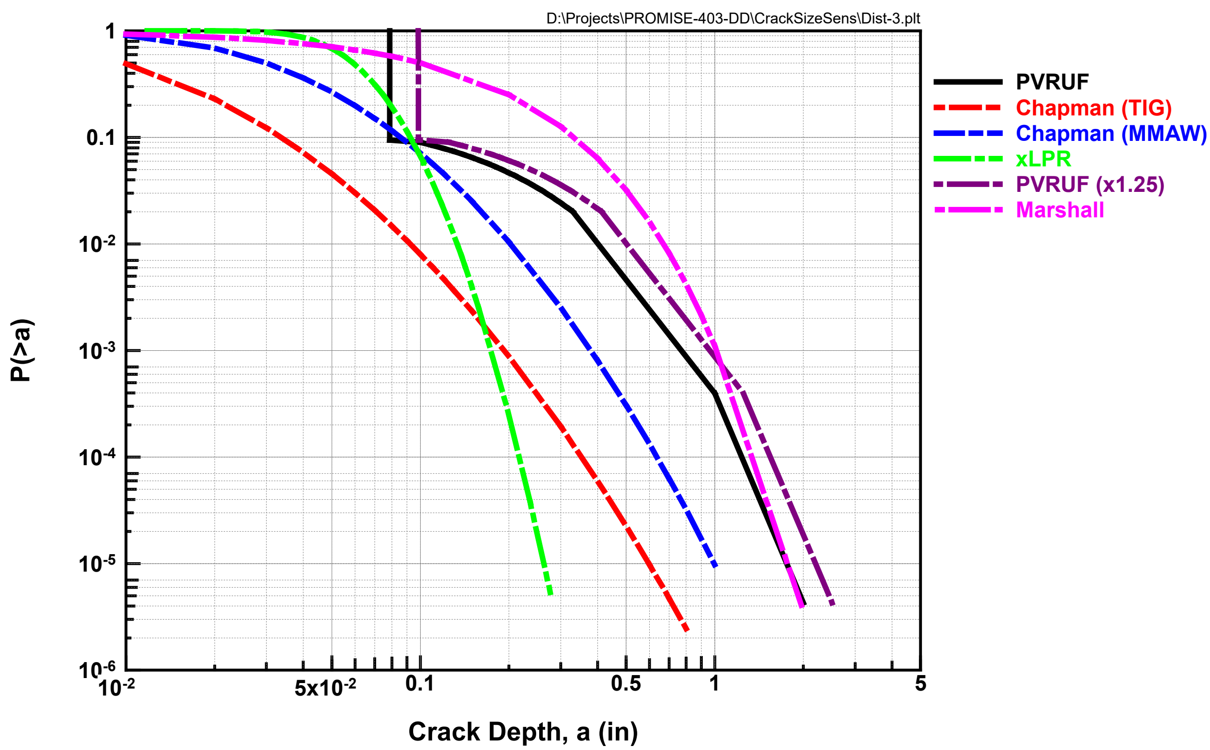

Figure RAI-4-1: Initial Flaw Size Distributions

PFM sensitivity study evaluations were performed with the six initial flaw size distributions for the limiting path in the pressurizer. Path PRSHC-BW-2C is limiting, as identified in Sections 8.3.4.1 and 8.3.4.1.1 (and corresponding Tables 8-10 and 8-11) of EPRI Report 3002015905.

The objective of the sensitivity evaluations is to determine the impact of these initial flaw size distributions and the sensitivity of their use on the PFM results. The Base Case scenario consisting of pre-service inspection (PSI) followed by inservice inspections (ISI) after 20, 40 and 60 years of operation (PSI+20+40+60), as summarized in Table 8-9 of EPRI Report 3002015905[(2)], was used in all of the sensitivity study cases.

The PFM results for the sensitivity studies are shown in Table RAI-4-1 for Path PRSHC-BW-2C.

Table RAI-4-1: Comparison of the Probabilities of Leakage and Rupture for Different Initial Flaw Size Distributions for Path PRSHC-BW-2C
(for Base Case = PSI+20+40+60)

Sensitivity Study Case	Probability of Rupture at 80 Years (per Year)	Probability of Leakage at 80 Years (per Year)
Flaw Distribution #1: Base Case using PVRUF *(same as Table 8-10 of EPRI Report 3002015905)*	1.25E-09	1.25E-09
Flaw Distribution #2: Base Case with PVRUF x 1.25	1.25E-09	6.25E-09
Flaw Distribution #3: Base Case with Marshall	1.25E-09	1.25E-09
Flaw Distribution #4: Base Case with Chapman, et al. (TIG)	1.25E-09	1.25E-09
Flaw Distribution #5: Base Case with xLPR	1.25E-09	1.25E-09
Flaw Distribution #6: Base Case with Constant 0.25t flaw	1.25E-09	1.25E-09

The results in Table RAI-4-1 indicate that the PFM results are insensitive to the initial flaw distribution. These results are consistent with the results of the sensitivity study documented in Section 8.3.4.3.6 of EPRI Report 3002015905, where the PVRUF distribution was compared to the Marshall and NUREG/CR-6817 flaw distributions using a stress multiplier of 1.80 (those results are shown in Table 8-26 of EPRI Report 3002015905). For the sensitivity study results summarized in Table RAI-4-1, all of the flaw distributions have the same probability of rupture after 80 years of operation ($1.25x10^{-9}$). Flaw Distribution #2 (PVRUF×1.25) has the highest probability of leakage ($6.25x10^{-9}$) after 80 years of operation, but this probability value is the same order of magnitude as the probabilities of leakage for the other four flaw distributions. All of the probabilities of rupture and leakage are well below the acceptance criterion of $1x10^{-6}$ per year.

Therefore, the assessment of the pressurizer welds in EPRI Report 3002015905 is acceptable because it used the PVRUF flaw distribution, and differences in geometry (wall thickness) relative to PVRUF or piping, differences in materials between vessels and piping, and differences in manufacturing processes relative to vessels and piping when assumed for the pressurizer all have a minor impact on the PFM results. As a result, the flaw distribution based

(2) Table 8-9 of EPRI Report 3002015905 incorrectly identifies the Base Case ISI as 20, 40, 60, and 80 years; however, as indicated elsewhere throughout the report (e.g., in the title for Table 8-10, etc.), the Base Case evaluated PSI+20+40+60. None of the PFM results in EPRI Report 3002015905 are affected by this error. EPRI will correct the error in Table 8-9 when the report is revised later this year.

on the PVRUF flaw data from an RPV is appropriate for application to the Salem Units 1 and 2 pressurizer lower head shell welds.

RAI 5

Regulatory Basis

The regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the requested PZR SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue

Page 5 of Attachment 1 to the submittal describes a sensitivity run performed for Salem with an examination coverage of 37% to address the 37.2% examination coverage achieved for weld 2-PZR-CIRC DUH during the 2nd 10-year ISI interval. It further states that this sensitivity run was made with preservice inspection (PSI) with five ISI inspections at 10, 20, 30, 40, and 70 years (PSI + 10 + 20 + 30 + 40 +70). The staff noted that the table of coverage on the same page of Attachment 1 to the submittal indicates that the fourth examination for weld 2-PZR-CIRC DUH has not yet been performed; therefore, the sensitivity run should be performed with PSI + 10 + 20 + 30 + 60. The staff also noted that only the probability of leakage (1.34x10-6) was reported and that it is not clear whether this probability of leakage is the value per year at 80 years. The staff noted the comparable probability of leakage value of 1.13x10-6 per year in Table 8-33 of EPRI report 3002015905, but it is not clear whether the Table 8-33 results are at 80 years. The staff also noted that only the probability of leakage values are reported in Table 8-33 of EPRI report 3002015905. In order to determine whether the low examination coverage achieved for weld 2-PZR-CIRC DUH during the 2nd 10-year ISI interval would result in a probability of rupture value below the criterion, the staff needs probability of rupture values per year at 80 years with the lower examination coverage.

Request

Either;

(a) Perform the 37% examination coverage run described on page 5 of Attachment 1 to the submittal except with PSI + 10 + 20 + 30 + 60, and report the resulting probability of rupture value per year at 80 years.

or

(b) For the limiting case (PRSHC-BW-2C) in Table 8-33 of EPRI report 3002015905, report the resulting probability of rupture value per year at 80 years for the base case with 50% examination coverage.

PSEG Response:

For the limiting Case ID PRSHC-BW-2C in Table 8-33 of EPRI Report 3002015905 for the Base Case (PSI+20+40+60) with 50% examination coverage (with a reported probability of leakage of 1.13E-06), the corresponding probability of rupture is 1.25E-09 per year after 80 years of operation.

RAI 6

Regulatory Basis

The regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the requested PZR SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is referenced, and thereby providing an acceptable level of quality and safety per10 CFR 50.55a(z)(1) for these welds.

Issue

Section 7.0 of Attachment 1 to the submittal states that the proposed alternative for Salem Unit 1 and 2 is requested for the remainder of the 4th inspection intervals through the 5th inspection intervals, currently scheduled to end on December 31, 2030. The Salem Unit 1 renewed operating license (ADAMS Accession No. ML052990140) states that the Salem Unit 1 renewed license shall expire on August 13, 2036. The Salem Unit 2 renewed operating license (ADAMS Accession No. ML052990143) states that the Salem Unit 2 renewed license shall expire on April 18, 2040. With these expiration dates of the renewed licenses, the 5th inspection interval for Salem Unit 1 should end on August 13, 2026; and the 5th inspection interval for Salem Unit 2 should end on April 18, 2030. The staff needs clarification on the end dates of the 5th inspection intervals for both units.

Request

Clarify if the 5th inspection interval for Salem Unit 1 ends on August 13, 2026; and if the 5th inspection interval for Salem Unit 2 ends on April 18, 2030, such that the proposed alternative would be requested up to August 13, 2026 for Salem Unit 1 and April 18, 2030 for Salem Unit 2.

PSEG Response:

Due to extended outages on both Salem Units 1 and 2 in the mid-1990s, the ISI 10-year Intervals are not in alignment with the Facility Operating License dates. Additionally the Unit 2 4^{th} ISI 10-Year interval was shortened to align with the end of the Unit 1 4^{th} ISI 10-year interval and the Containment (CISI) 10-year intervals. The 5^{th} ISI 10-year intervals for both Salem Unit 1 and 2 are scheduled to end on December 31, 2030 as documented in the ISI program plan. The 6^{th} 10-year ISI intervals would extend to the end of the current period of extended operation of August 13, 2036 for Unit 1 and April 18, 2040 for Unit 2 or to December 31, 2040 if subsequent license renewal is pursued.

The proposed 30 year examinations would begin to be scheduled during the 6^{th} 10-year ISI intervals and if applicable beyond the current period of extended operation.

RAI 7

Regulatory Basis

The NRC has established requirements in 10 CFR Part 50 to protect the structural integrity of structures and components in nuclear power plants. Among these requirements are the ISI requirements of Section XI of the ASME Code incorporated by reference in 10 CFR Part 50.55a to ensure that adequate structural integrity of the PZR vessel is maintained through the service life of the vessel. Therefore, the regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the PZR SHWs of Salem Units 1 and 2 for which EPRI report 3002015905 is

referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these components.

Issue
The NRC staff noted that the probabilistic fracture mechanics (PFM) software used in EPRI report 3002015905 is PROMISE Version 2.0. The NRC staff audited PROMISE Version 1.0 and issued the audit report by letter dated December 10, 2020 (ADAMS Accession No. ML20258A002). One of the objectives of the NRC staff's audit was to ensure that PROMISE Version 1.0 received adequate verification and validation (V&V). The NRC staff has not audited PROMISE Version 2.0. 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."

Request
To ensure that PROMISE Version 2.0 received adequate V&V for use in the referenced analyses for the PZR SHWs of Salem Units 1 and 2, describe the V&V performed for PROMISE Version 2.0 to ensure that (1) cases intended for the difference between PROMISE Version 2.0 and PROMISE Version 1.0 reflected the change; and (2) that cases not intended for the difference were not affected.

PSEG Response:

The usage of two different versions of the PROMISE software is due to the timing of the publication of the two EPRI reports and improvements made to the software between the publication dates. EPRI Report 3002014590 (for SG Class 2 pressure-retaining welds) was published in April 2019 and EPRI Report 3002015905 (for pressurizer welds) was published in December 2019. PROMISE Version 2.0 was developed in May 2019. The NRC performed their audit of Version 1.0 of the PROMISE software because that version of the software was used in another plant-specific submittal that was only based on the earlier EPRI Report 3002014590.

The main difference between the two versions of the PFM software is that PROMISE Version 1.0 applies a single, user-specified examination coverage to all inspections assumed over the component evaluation time period, whereas PROMISE Version 2.0 applies unique, user-specified examination coverages to each inspection assumed over the component evaluation period. In both PROMISE Versions 1.0 and 2.0, the software assumes 100% coverage for the pre-service inspection (PSI) examination.

Structural Integrity Associates performed a V&V and documented the results in a V&V calculation package. With respect to Item (1) of the NRC’s request, test cases were run that were specific to testing the portions of the software that were modified. Table RAI-7-1, which has been reproduced from the V&V package, contains the results for four scenarios where the coverages for each inspection during the component lifetime were not all identical. Besides inspection coverage, all other PFM inputs were identical for all four V&V scenarios. For V&V Scenario A, which was considered to be the reference case, the coverage was assumed to be 100% for the PSI at 0 years, followed by inspection coverage values of 50% for all remaining examinations at 20, 40, and 60 years of operation. The other three V&V scenarios assumed different inspection coverage values for 20, 40, and 60 years.

As shown by the results in Table RAI-7-1, there is a notable difference in the cumulative probability of leakage at 80 years for Scenario B compared to Scenario A. These results were verified to be correct in the V&V. The results for Scenarios C and D are the same as Scenario A because early inspections during the life of the component have the most impact in reducing the probability of leakage, while PFM sensitivity studies have shown that later inspections do not have a significant impact on the probability of leakage. This is demonstrated by the sensitivity study results presented in Table 8-30 of EPRI Report 3002015905 in which an inspection scenario of (0, 10, 20 and 30 years) resulted in a probability of failure of 2.50E-09, while an inspection scenario of (0, 10, 20, 30, 50 and 70 years) resulted in a probability of failure of 1.25E-09 using the BWRVIP-108 probability of detection (POD) curve or 2.50E-09 using the Appendix L POD curve. Similar results were obtained for EPRI Report 3002014590 as documented in the response to RAI 3 for the Vogtle Electric Generating Plant, Units 1 & 2 (ADAMS Accession No. ML20329A302 dated November 23, 2020, Response to RAI 3 and Attachment 1, “Response to Open Audit Item 2.c.i.B”). These results were verified to be correct in the V&V.

Finally, it is noted there were no failures due to ruptures in all of the V&V scenarios, so only the probability of leakage results are meaningful for these scenarios.

Based on the foregoing evaluation, these V&V test cases verified that the user-input inspection coverage was implemented correctly in PROMISE Version 2.0.

Table RAI-7-1: 80-Year Probability of Leakage Values for Various Assumed Inspection Coverage Scenarios

V&V Scenario	Coverage Values for Inspections at 0, 20, 40, 60 Years (%)	Probability of Leakage at 80 Years
A (reference case)	100, 50, 50, 50	0.149
B	100, 0, 50, 50	0.1686
C	100, 50, 0, 50	0.149
D	100, 50, 50, 0	0.149

With respect to Item (2) of the NRC's request, the V&V for PROMISE Version 2.0 performed some of the same test runs that were performed in the V&V for PROMISE Version 1.0. Table RAI-7-2, which was reproduced from the V&V package, contains the results of tests where the output from PROMISE Version 2.0 is compared with the output from PROMISE Version 1.0 for four cases where assumed inspection coverage is identical for all inspections during the component lifetime. Besides inspection coverage, other PFM inputs were identical for all four test runs. As indicated in this table, the cumulative probability of leakage results are the same for both versions of the software. There were no failures due to ruptures in all cases, so only the probability of leakage results are reported.

Furthermore, all the test cases for PROMISE Version 1.0 that investigated stress intensity factors, crack growth laws, crack size distributions and random sampling were re-run in PROMISE Version 2.0. The results of those cases were identical, and no discrepancies were observed. This assures that no differences other than those intended were introduced in PROMISE Version 2.0.

Table RAI-7-2: 80-Year Probability of Leakage Values for PROMISE Versions 1.0 and 2.0 for Various Assumed Inspection Coverage Scenarios

Inspection Coverage Assumed at 20, 40, and 60 Years (%)	PROMISE Version 1.0 Results for One Crack	PROMISE Version 2.0 Results for One Crack
0	0.296	0.296
25	0.223	0.223
50	0.149	0.149
100	8.52E-04	8.52E-04

Therefore, the V&V testing performed for PROMISE Version 2.0 demonstrated that: (1) the results for the cases intended to test the differences between PROMISE Version 2.0 and PROMISE Version 1.0 correctly reflected the software changes, and (2) the results for the cases re-run from the PROMISE Version 1.0 testing were not affected.

References:

1. PSEG Letter LR-N20-0041 to NRC, “Proposed Alternative for Examination of ASME Section XI, Examination Category B-B, Item Number B2.11 and B2.12,” dated August 5, 2020