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10 CFR 50
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Nuclear Generating Station, Unit No. 1 and Unit No. 2
Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: Response to NRC Request for Additional Information dated March 22, 2010, Related to Section 4 of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application

Reference: Letter from Mr. Donnie Ashley (USNRC) to Mr. Thomas Joyce (PSEG Nuclear, LLC) "REQUEST FOR ADDITIONAL INFORMATION RELATED TO SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION SECTION 4.2, "REACTOR VESSEL NEUTRON EMBRITTLEMENT," SECTION 4.4.1, "REACTOR VESSEL UNDERCLAD CRACKING ANALYSES," AND SECTION 4.4.2, "REACTOR COOLANT PUMP FLYWHEEL FATIGUE CRACK GROWTH ANALYSES," dated March 22, 2010

In the referenced letter, the NRC requested additional information related to Section 4 of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application (LRA). Enclosed are the responses to this request for additional information.

This letter and its enclosure contain no regulatory commitments.

If you have any questions, please contact Mr. Ali Fakhar, PSEG Manager - License Renewal, at 856-339-1646.

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on 4/20/10

Sincerely,

Paul J. Davison

Paul J. Davison
Vice President, Operations Support
PSEG Nuclear LLC

Enclosure: Responses to Request for Additional Information

cc: S. Collins, Regional Administrator – USNRC Region I
B. Brady, Project Manager, License Renewal – USNRC
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector – Salem
P. Mulligan, Manager IV, NJBNE
L. Marabella, Corporate Commitment Tracking Coordinator
Howard Berrick, Salem Commitment Tracking Coordinator

Enclosure

Responses to Request for Additional Information related to Section 4 of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application (LRA)

RAI 4.2.1-1
RAI 4.2.2-1
RAI 4.2.2-2
RAI 4.2.3-1
RAI 4.4.2-1

Note : In the responses, impacts to the LRA are explained, and where appropriate to facilitate understanding, portions of the LRA are repeated with the change highlighted by strikethroughs for deleted text and bolded italics for inserted text.

RAI 4.2.1-1

License renewal application (LRA) Section 4.2.1, "Neutron Fluence Analyses," stated "The current reactor vessel embrittlement analyses that evaluate reduction of fracture toughness of the Salem Nuclear Generating Station, Units 1 and 2 (Salem) reactor vessel beltline materials are based on predicted 40-year end-of-license (EOL) fluence values of 32 Effective Full Power Years (EFPY)."

Please confirm that the current licensing basis 32 EFPY reactor vessel embrittlement analyses for Salem, Units 1 and 2 are those approved in a safety evaluation (SE) dated May 25, 2001, regarding a power uprate request. This request included a pressure-temperature (P-T) limits revision and an exemption request to use American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves." WCAP-15565, Revision (Rev.) 1, "Salem Unit 1 Heatup and Cooldown Curves for Normal Operation," and WCAP-15566, Rev. 1, "Salem Unit 2 Heatup and Cooldown Curves for Normal Operation," are supplements to support the review of the P-T limits revision. Both are dated February 2001 and contain the 32 EFPY and the 48 EFPY fluence values for Salem reactor pressure vessel (RPV) materials.

Please (1) provide basis for reducing the RPV fluence value from 2.42E+19 at 48 EFPY (WCAP-15565, Rev. 1) to 1.83E+19 at 50 EFPY (LRA) for Unit 1 and from 2.66E+19 at 48 EFPY (WCAP-15566, Rev. 1) to 1.96E+19 at 50 EFPY (LRA) for Unit 2, and (2) supplement LRA Section 4.8, "References," by providing additional list of references from which the fluence values in LRA Tables 4.2.1-1 (Unit 1) and 4.2.1-2 (Unit 2) were obtained.

PSEG Response:

Confirmation of CLB

The Staff approved PSEG's submittal for a license amendment from 3,411 MWt to 3,459 MWt in their SER dated May 25, 2001 (ADAMS Accession No. ML011350051). The SER referenced WCAP-15565, Rev. 1, "Salem Unit 1 Heatup and Cooldown Curves for Normal Operation," and WCAP-15566, Rev. 1, "Salem Unit 2 Heatup and Cooldown Curves for Normal Operation," as supplements to the submittal. The 32 EFPY reactor vessel embrittlement analyses for Salem 1 and Salem 2 are those supporting the current licensing basis which was approved in the SER.

Basis for Reduction of RPV Fluence Values

The Salem Unit 1 Reactor Pressure Vessel (RPV) surface fluence values for the Lower Shell Plates B2403-1, B2403-2, B2403-3, and the Lower to Intermediate Shell Circumferential Weld were determined to be 2.42E+19 n/cm^2 (E> 1.0 MeV) at 48 EFPY in Table 4-2 of WCAP-15565, Rev. 1. The Salem LRA Table 4.2.1-1 fluence values for these same components were determined to be 1.83E+19 n/cm^2 (E> 1.0 MeV) at 50 EFPY, which are lower than the previous values.

The Salem Unit 2 RPV surface fluence values for the Lower Shell Plates B4713-1, B4713-2, B4713-3, and the Lower to Intermediate Shell Circumferential Weld were determined to be 2.66E+19 n/cm^2 (E> 1.0 MeV) at 48 EFPY in Table 4-2 of WCAP-15566, Rev. 1. The Salem

LRA Table 4.2.1-2 fluence values for these same components were determined to be 1.96E+19 n/cm^2 (E> 1.0 MeV) at 50 EFPY, which are lower than the previous values.

There are several factors that resulted in the changes in projected reactor vessel fluence between the data published in WCAP-15565, Rev. 1 and WCAP-15566, Rev. 1 and Tables 4.2.1-1 and 4.2.1-2 of the Salem LRA. These factors are summarized below.

1. Fluence Methodology Changes

   The fluence data provided in WCAP-15565, Rev. 1 and WCAP-15566, Rev.1 were generated with the approved fluence methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Rev. 2, January 1996.

   The fluence results provided in Salem LRA Tables 4.2.1-1 and 4.2.1-2 were generated with the updated and approved fluence methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Rev. 4, May 2004 (ADAMS Accession No. ML040620297) and WCAP-16083-NP-A, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry", Rev. 0, May 2006 (ADAMS Accession No. ML061600256).

2. Incorporation of Actual Power Uprate Implementation

   In the fluence results provided in WCAP-15565, Rev. 1 and WCAP-15566, Rev. 1, the impact of a proposed power uprate from 3411 MWt to 3459 MWt was conservatively accounted for by assuming that the power uprate occurred coincident with the withdrawal of the last (relative to the analysis date) surveillance capsule.

   This assumption placed the uprate at the onset of Cycle 13 for Salem Unit 1 and Cycle 7 for Salem Unit 2. This conservatism was removed in the LRA analysis by incorporating the actual dates of the power uprate into the analysis. Thus, the Salem LRA fluence evaluations accounted for actual uprate occurring at the onset of Cycle 15 for Unit 1 and near the middle of Cycle 12 for Unit 2. The net effect of this was to reduce the incremental fluence for Cycles 13 and 14 at Salem Unit 1 and for Cycles 7 through the middle of 12 at Salem Unit 2.

3. Use of Actual Fuel Cycle Analyses

   The fluence analyses documented in WCAP-15565, Rev. 1 and WCAP-15566, Rev. 1 included fuel cycle specific analysis for Cycles 1 through 12 at Salem Unit 1 and Cycles 1 through 6 at Salem Unit 2. Fluence projections beyond those cycles were based on the continued use of fuel cycle designs characteristic of those being implemented at that time.

   The fluence results documented in the Salem LRA Tables 4.2.1-1 and 4.2.1-2 included fuel cycle specific analysis for Cycles 1 through 18 at Salem Unit 1 and Cycles 1 through 15 at Salem Unit 2. The updated analysis for the Salem LRA, therefore, replaced 6 cycles of projected fluence for Salem Unit 1 with actual cycle specific results and 9

cycles of projected fluence for Salem Unit 2 with actual cycle specific results, which had the effect of reducing the fluence values as shown in LRA Tables 4.2.1-1 and 4.2.1-2 from those that were projected in WCAP-15565, Rev. 1, and WCAP-15566, Rev. 1.

4. Implementation of Low Leakage Loading Patterns

Over the operating lifetime of Salem Units 1 and 2, the fuel cycle designs have transitioned to low leakage fuel management. As a result, the incident neutron flux at the pressure vessel inner diameter has likewise been reduced and the neutron fluxes used for the LRA fluence projections beyond Cycle 18 at Unit 1 and beyond Cycle 15 at Unit 2 are significantly reduced from those that were used for the fluence projections listed in WCAP-15565, Rev. 1 and WCAP-15566, Rev. 1.

The above four improvements to the fluence analyses (fluence methodology changes, incorporation of actual power uprate implementation, use of actual fuel cycle analyses, and implementation of low leakage loading patterns) result in the overall neutron fluence reductions noted in the LRA relative to the projections documented in the WCAP-15565, Rev. 1, and WCAP-15566, Rev. 1, reports.

References for Fluence Values

The references from which the fluence values in LRA Tables 4.2.1-1 and 4.2.1-2 were obtained are listed below. LRA Section 4.8, "References," is updated as follows:

4.8.7 WCAP-16981-NP, Rev. 1, "Salem Unit 1 Time-Limited Aging Analysis on Reactor Vessel Integrity"

4.8.8 WCAP-16982-NP, Rev. 0, "Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity"

RAI 4.2.2-1

LRA Section 4.2.2, "Upper Shelf Energy [(USE)] Analyses," states that Charpy USE for the beltline forgings and welds of Salem, Units 1 and 2 were determined using surveillance data and the Charpy USE for the RPV extended beltline materials was determined without the use of surveillance data. This statement is not consistent with information in LRA Table 4.2.2-1 for Unit 1, which shows that surveillance data was used for evaluating only the intermediate shells, and information in LRA Table 4.2.2-2 for Unit 2, which shows that surveillance data was used for evaluating only one intermediate shell.

The Nuclear Regulatory Commission's (NRC's) Reactor Vessel Integrity Database (RVID) indicates that, in addition to the intermediate shells, Intermediate Shell Axial Weld 2-042 of the Unit 1 RPV has more than one surveillance data point; likewise, WCAP-15692, "Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," indicates that, in addition to Intermediate Shell B4712-2, Intermediate Shell Axial Weld 2-442 of the Unit 2 RPV has more than one surveillance data. For these weld materials having at least two surveillance data, please use Position 2.2 of the Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," to evaluate their USE values and revise the subject statement appropriately. Note that there are no criteria in RG 1.99, Rev. 2 to determine credibility of measured USE data.

PSEG Response:

The third sentence in the first complete paragraph on Salem LRA Page 4-11 reads as follows:

> Charpy USE for the beltline forgings and welds were determined using surveillance data (Position 2.2 of the Regulatory Guide), and the Charpy USE for the extended beltline materials was determined without the use of surveillance data (Position 1.2 of the Regulatory Guide).

The above statement in the LRA is inconsistent with LRA Tables 4.2.2-1 and 4.2.2-2. In order to correct the inconsistency and provide further clarification, the LRA sentence cited above is replaced as follows:

> ***The Charpy USE values for the beltline and extended beltline plates and welds were determined using Figure 2 of Regulatory Guide 1.99, Revision 2, and are listed in Salem LRA Tables 4.2.2-1 and 4.2.2-2. Where two or more credible surveillance data sets were available, Position 2.2 of the Regulatory Guide was used to determine predicted Charpy USE values as listed in Salem LRA Tables 4.2.2-1 and 4.2.2-2. The use of surveillance capsule data (Position 2.2 of the Regulatory Guide) was applied to all three Intermediate Shell plates B2402-1, B2402-2, and B2402-3 for Salem Unit 1 and Intermediate Shell Plate B4712-2 for Salem Unit 2.***
>
> ***For the remaining plates and welds within the beltline and extended beltline regions, Position 1.2 of the Regulatory Guide was used to determine predicted Charpy USE values. The remaining plates do not have surveillance capsule data.***

The welds have surveillance capsule data, however, the surveillance weld material in the capsules represents only one of the two heats contained in the vessel welds, thus are not identical to those materials used in the fabrication of the welds of the respective Salem Units 1 and 2 vessels. Therefore, the surveillance capsule data was not used in determining the predicted USE values.

The basis of the revised statement is provided as follows: Although the NRC's RVID indicates that the Intermediate Shell Axial Weld 2-042 of the Unit 1 RPV has more than one surveillance data point, Salem had selected Regulatory Guide 1.99, Rev. 2, Position 1.2 to determine the decrease in Upper Shelf Energy (USE) values for the Intermediate Shell Longitudinal (Axial) Welds 2-042 A, B, and C. The surveillance weld was fabricated from weld wire heat 39B196. The intermediate shell longitudinal weld was fabricated using weld wire heats 39B196 and 34B009 with Ni-200. Since the surveillance weld material only represents one of the two heats contained in the vessel weld, the most accurate prediction of the USE decrease is using Position 1.2 where the decrease in USE is predicted based on the copper content of the vessel weld.

For Salem Unit 2, although WCAP-15692 indicates that the Intermediate Shell Axial Weld 2-442 of the Unit 2 RPV has more than one surveillance data and also deemed this data credible, Salem had selected Regulatory Guide 1.99, Rev. 2, Position 1.2 to determine the decrease in USE values for the Intermediate Shell Longitudinal (Axial) Welds 2-442 A, B, and C. The surveillance weld is fabricated from weld wire heat 13253. The intermediate shell longitudinal weld is fabricated using weld wire heats 13253 and 20291. Since the surveillance weld material only represents one of the two heats contained in the vessel weld, the most accurate prediction of the USE decrease is using Position 1.2 where the decrease in USE is predicted based on the copper content of the vessel weld.

RAI 4.2.2-2

Unlike LRA Tables 4.2.2-1 and 4.2.2-2, the Nuclear Regulatory Commission's (NRC) Reactor Vessel Integrity Database (RVID) does not contain information for the extended beltline materials of the Salem, Units 1 and 2 RPVs. Please discuss the procedures that you used to determine the chemistry data, initial reference temperature (RTNDT), margins and initial USE values for the extended beltline materials to demonstrate that you have applied consistent approaches in determining the above mentioned material information for both beltline and extended beltline materials.

PSEG Response:

Salem Unit 1

Salem applied a consistent approach in determining chemistry data, initial reference temperature (RT_{NDT}), margins, and initial USE values for both the extended beltline materials and the beltline materials of the Unit 1 reactor vessel.

The process for determining the metal embrittlement properties of the extended beltline materials is as follows. As listed in Salem License Renewal Application (LRA) Table 4.2.1-1, "50 EFPY Surface Fluence Projections for Beltline and Extended Beltline Materials for Salem Unit 1", the following materials are considered extended beltline; Upper Shell Plates, Upper Shell Longitudinal Weld Seams, and the Intermediate to Upper Shell Circumferential Weld Seam.

Upper Shell Plates B2401-1, B2401-2, and B2401-3

The percentage by weight of copper (wt% Cu) for the upper shell plates is not recorded in the Certified Material Test Reports (CMTR) for Salem Unit 1 reactor vessel. Per the CMTR, the material for the upper shell plates is ASTM A302B Modified, where the ASTM A302B material specification did not require analysis of copper content. 10 CFR 50.61 indicates that an analysis may be used to determine a generic value of wt% Cu when no information is otherwise available. Chemistry measurements are available for ASTM A302B Modified in the Embrittlement Shift Transition Temperature Shift (TTS) Database that is attached to the Oak Ridge National Laboratory Report ORNL/TM-2006/530. Appendix G of the ORNL report tabulates the mean, standard deviation, and maximum values for the various material specifications. The Salem Unit 1 calculations utilized the maximum copper content for A302B Modified materials from the ORNL report, which is 0.25 wt% for RT_{NDT} calculations of the Salem Unit 1 upper shell plates (B2401-1, B2401-2, B2401-3).

The Upper Shell Plates percentage by weight nickel (wt% Ni) values were taken from the CMTRs.

The initial Reference Temperature (RT_{NDT}) values were based on the CMTRs and were determined in accordance with subparagraph NB-2331 of ASME Code Section III.

The margins were calculated using the methodology presented in Regulatory Guide 1.99, Rev. 2. Following guidance in WCAP-14040-A, the value of σ_i was 0 °F since the initial RT_{NDT} values were measured. Per the Regulatory Guide, the value of σ_Δ used in the margin calculation was determined to be 17°F for base metal; however, σ_Δ need not exceed 0.5 times the ΔRT_{NDT} mean value.

The Salem Unit 1 Upper Shell Plate initial USE values were calculated based on ASTM E185-82 and Charpy test data points in the CMTRs.

Upper Shell Longitudinal Weld Seams 1-042 A, B, and C

The wt% Cu and wt% Ni values were taken from Combustion Engineering Report CE NPSD-1119, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", Rev. 1, dated July 1998. Please note that this CE NPSD-1119 report is the updated version of the NRC reviewed report, CE-NPSD-1039, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", Rev. 2, dated June 1997 (ADAMS Accession No. 9707180236).

The initial Reference Temperature (RT_{NDT}) values are generic values taken from 10 CFR 50.61.

The margins were calculated using the methodology presented in Regulatory Guide 1.99, Rev. 2. Following guidance in 10 CFR 50.61, the value of σ_i was 17 °F since the initial RT_{NDT} values are generic mean values. Per the Regulatory Guide, the value of σ_Δ used in the margin calculation was determined to be 28°F for welds; however, σ_Δ need not exceed 0.5 times the ΔRT_{NDT} mean value.

Generic USE values were based on CEN-622-A Final Report, "Generic Upper Shelf Values for Linde 1092, 124, and 0091 Reactor Vessel Welds, CEOG Task 839," December 1996.

Intermediate to Upper Shell Circumferential Weld Seam 8-042

The wt% Cu and wt% Ni values were taken from Combustion Engineering Report CE NPSD-1119, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", Rev. 1, dated July 1998. Please note that this CE NPSD-1119 report is the updated version of the NRC reviewed report, CE-NPSD-1039, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", Rev. 2, dated June 1997 (ADAMS Accession No. 9707180236).

The initial Reference Temperature (RT_{NDT}) values are generic values taken from 10 CFR 50.61.

The margins were calculated using the methodology presented in Regulatory Guide 1.99, Rev. 2. Following guidance in 10 CFR 50.61, the value of σ_i was 17 °F since the initial RT_{NDT} values are generic mean values. Per the Regulatory Guide, the value of σ_Δ used in the margin calculation was determined to be 28°F for welds; however, σ_Δ need not exceed 0.5 times the ΔRT_{NDT} mean value.

Generic USE values were based on CEN-622-A Final Report, "Generic Upper Shelf Values for Linde 1092, 124, and 0091 Reactor Vessel Welds, CEOG Task 839," December 1996.

Salem Unit 2

Similarly for Salem Unit 1, a consistent approach was applied in determining chemistry data, initial reference temperature (RT_{NDT}), margins, and initial USE values for both the extended beltline materials and the beltline materials of the Unit 2 reactor vessel.

The process for determining the metal embrittlement properties of the Salem Unit 2 extended beltline materials is as follows. As listed in Salem LRA Table 4.2.1-2, "50 EFPY Surface Fluence Projections for Beltline and Extended Beltline Materials for Salem Unit 2", the following materials are considered extended beltline; Upper Shell Plates, Upper Shell Longitudinal Weld Seams, and the Intermediate to Upper Shell Circumferential Weld Seam.

Upper Shell Plates B4711-1, B4711-2, and B4711-3

The Upper Shell Plates percentage by weight copper (wt% Cu) and nickel (wt% Ni) values were taken from the CMTRs.

The initial Reference Temperature (RT_{NDT}) values were based on the CMTRs and were determined in accordance with subparagraph NB-2331 of ASME Code Section III.

The margins were calculated using the methodology presented in Regulatory Guide 1.99, Rev. 2. Following guidance in WCAP-14040-A, the value of σ_i was 0 °F since the initial RT_{NDT} values were measured. Per the Regulatory Guide, the value of σ_Δ used in the margin calculation was determined to be 17°F for base metal; however, σ_Δ need not exceed 0.5 times the ΔRT_{NDT} mean value.

The Salem Unit 2 Upper Shell Plate initial USE values were calculated based on ASTM E185-82 and Charpy test data points in the CMTRs.

Upper Shell Longitudinal Weld Seams 1-442 A, B, and C

The wt% Cu and wt% Ni values were taken from Combustion Engineering Report CE NPSD-1119, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", Rev. 1, dated July 1998. Please note that this CE NPSD-1119 report is the updated version of the NRC reviewed report, CE-NPSD-1039, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", Rev. 2, dated June 1997 (ADAMS Accession No. 9707180236).

The initial Reference Temperature (RT_{NDT}) values are generic values taken from 10 CFR 50.61.

The margins were calculated using the methodology presented in Regulatory Guide 1.99, Rev. 2. Following guidance in 10 CFR 50.61, the value of σ_i was 17 °F since the initial RT_{NDT} values are generic mean values. Per the Regulatory Guide, the value of σ_Δ

used in the margin calculation was determined to be 28°F for welds; however, σ_Δ need not exceed 0.5 times the ΔRT_{NDT} mean value.

Generic USE values were based on CEN-622-A Final Report, "Generic Upper Shelf Values for Linde 1092, 124, and 0091 Reactor Vessel Welds, CEOG Task 839," December 1996.

Intermediate to Upper Shell Circumferential Weld Seam 8-442

The wt% Cu and wt% Ni values were taken from Combustion Engineering Report CE NPSD-1119, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", Rev. 1, dated July 1998. Please note that this CE NPSD-1119 report is the updated version of the NRC reviewed report, CE-NPSD-1039, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", Rev. 2, dated June 1997 (ADAMS Accession No. 9707180236).

The initial Reference Temperature (RT_{NDT}) values are generic values taken from 10 CFR 50.61.

The margins were calculated using the methodology presented in Regulatory Guide 1.99, Rev. 2. Following guidance in 10 CFR 50.61, the value of σ_i was 17 °F since the initial RT_{NDT} values are generic mean values. Per the Regulatory Guide, the value of σ_Δ used in the margin calculation was determined to be 28°F for welds; however, σ_Δ need not exceed 0.5 times the ΔRT_{NDT} mean value.

Generic USE values were based on CEN-622-A Final Report, "Generic Upper Shelf Values for Linde 1092, 124, and 0091 Reactor Vessel Welds, CEOG Task 839," December 1996.

RAI 4.2.3-1

In LRA Tables 4.2.3-1 and 4.2.3-2, the applicant presented the pressurized thermal shock (PTS) reference temperature, RT_{PTS}, values for 50 EFPY for Salem, Units 1 and 2. Also presented in these tables are input parameters necessary for calculating the RT_{PTS} values. The staff found some discrepancies between these LRA Tables and those in the RVID, or WCAP-15565, Rev. 1, and requests the applicant to address them:

Unit 1

- LRA Table 4.2.3-1 shows that the chemistry factors based on surveillance data of intermediate shell plates, B2402-1, B2402-2, and B2402-3 are different from those documented in RVID, or WCAP-15565, Rev. 1. Provide basis for your revision.

- LRA Table 4.2.3-1 shows that the chemistry data for the lower shell longitudinal weld 3-042C (the limiting beltline material of Salem, Unit 1) is different from those documented in WCAP-15565, Rev. 1 which is consistent with the RVID data. Provide basis for your revision.

Unit 2

- LRA Table 4.2.3-2 shows a chemistry factor of 189.1 °F based on the table of RG 1.99, Rev. 2 for Intermediate Shell Longitudinal Weld 2-442. However, WCAP-15692 shows a value of 194.53 °F based on surveillance data for this weld. Justify your approach of not using surveillance data to estimate the chemistry factor for this weld.

PSEG Response:

Unit 1

Chemistry Factors (CF) Differences for Intermediate Shell Plates B2402-1, B2402-2, and B2402-3

The current licensing basis (CLB) CF values were previously documented in Table 4-7, "Summary of the Salem Unit 1 Reactor Vessel Beltline Material Chemistry Factors Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1" of WCAP-15565, Rev. 1, "Salem Unit 1 Heatup and Cooldown Curves for Normal Operation". The CF values were originally calculated using both Positions 1.1 and 2.1 of Regulatory Guide 1.99, Rev. 2.

The CF values contained in Salem LRA Table 4.2.3-1, "Calculation of RT_{PTS} Values for 50 EFPY at the Clad/Base Metal Interface for Salem Unit 1" are only based on Position 2.1 of Regulatory Guide 1.99, Rev. 2. The Position 2.1 CF values in LRA Table 4.2.3-1 are slightly higher as compared to the Position 2.1 CF values presented in Table 4-7 of WCAP-15565, Rev. 1, due to the use of updated surveillance capsule fluence values, which were re-evaluated as part of the Time-Limited Aging Analyses (TLAA) effort.

Chemistry Data Differences for Lower Shell Longitudinal Weld 3-042C

Salem LRA Table 4.2.3-1 contains wt% Cu and wt% Ni values for Unit 1 Lower Shell Longitudinal Weld Seams 3-042 A, B, and C that are different from those in the RVID and in WCAP-15565, Rev. 1. Since the Lower Shell Longitudinal Weld Seams are limiting, and are approaching the 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening criteria at 50 EFPY, wt% Cu and wt% Ni values were taken from an alternate source of information, ATI Consulting Report "A Review of Materials Data for the Salem Unit 1 Reactor Pressure Vessel Final Report", dated May 17, 1995. All of the Salem Unit 1 Reactor Vessel Lower Shell Longitudinal Weld Seams (3-042 A, B, and C) are comprised of Heat # 34B009+Ni-200; therefore, this response addresses all three weld seams. For Heat # 34B009+Ni-200, the copper and nickel weight percentage (wt% Cu and wt% Ni) data from Tables 11 and 12 of the ATI Report were averaged to be 0.1826 and 0.9825, respectively.

During preparation of this RAI response, a more complete data set was found in Combustion Engineering Report CE NPSD-1119, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content", Rev. 1, dated July 1998. Please note that this CE NPSD-1119, Rev. 1 report is the updated version of the NRC reviewed report, CE NPSD-1039, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", Rev. 2, dated June 1997 (ADAMS Accession No. 9707180236). It was concluded that the wt% Cu and wt% Ni values contained in CE NPSD-1119, Rev. 1 for Heat # 34B009+Ni-200 (0.192 and 1.007, respectively) are the most inclusive and accurate data. Therefore, these values for wt% Cu and wt% Ni are now used in the TLAA calculations for the Unit 1 Lower Shell Longitudinal Weld Seams 3-042 A, B, and C.

Salem has re-analyzed the 50 EFPY Charpy USE determinations for the Unit 1 Lower Shell Longitudinal Weld Seams 3-042 A, B, and C using the wt% Cu and wt% Ni values from the CE NPSD-1119, Rev. 1 report, which provides the best estimate wt% Cu and wt% Ni values as 0.192 and 1.007, respectively for Heat # 34B009+Ni-200.

As a result of using the wt% Cu and wt% Ni values from the CE NPSD-1119, Rev. 1 report, the 50 EFPY Charpy USE values were re-calculated using the same methodology as presented in WCAP-15565, Rev. 1. Due to the revision to the wt% Cu and wt% Ni values, LRA Table 4.2.2-1 is revised as shown below.

Table 4.2.2-1 Predicted USE Values at 50 EFPY for Salem Unit 1

RPV Material	Cu (%)	1/4T Fluence (E+19 n/cm^2)	Initial USE (ft-lb)	USE Decrease (%)	USE (ft-lb)
Lower Shell Longitudinal Weld Seams 3-042 A & B	~~0.1826~~ ***0.192***	0.668	112	31	77
Lower Shell Longitudinal Weld 3-042C	~~0.1826~~ ***0.192***	1.091	112	35	73

As seen from the updates made in the above Table, the 50 EFPY Charpy USE values for Lower Shell Longitudinal Weld Seams A, B, and C did not change and still remain greater than the 50

ft-lb criteria in Appendix G of 10 CFR 50. The slight increase in wt% Cu from 0.1826 to 0.192 results in a zero net change in USE decrease due to conservatively rounding up to the next wt% Cu line on Figure 2 of Regulatory Guide 1.99, Rev. 2.

Additionally, as a result of using the wt% Cu and wt% Ni values from the CE NPSD-1119, Rev. 1 report, the Chemistry Factor (CF), the shift in Reference Temperature (ΔRT_{PTS}), Margin, and Pressurized Thermal Shock Reference Temperature (RT_{PTS}) were re-calculated for all three weld seams using the same methodology as presented in WCAP-15567, Rev. 0. Due to the revision of the wt% Cu, wt% Ni, CF, ΔRT_{PTS}, and RT_{PTS} values, LRA Table 4.2.3-1 is revised as shown below.

Note in this table the Heat for Lower Shell Longitudinal Weld Seams 3-042 A, B & C was corrected to refer to "34B009+Ni-200, Linde 1092, Lot 3708", thereby, deleting reference to 13253. This is now consistent with the RVID. The weld chemistries used for both Salem License Renewal and for the previously approved WCAP-15565, Rev. 1, were based only on 34B009+Ni-200, Linde 1092, Lot 3708, and did not include 13253.

Table 4.2.3-1 Calculation of RT_{PTS} Values for 50 EFPY at the Clad/Metal Interface for Salem Unit 1

RV Beltline Region Location	Heat	Cu (%)	Ni (%)	Chem Factor (°F)	Surface Fluence (E+19 *n*/cm^2)	ΔRT_{PTS}	$RT_{NDT(U)}$ (°F)	Margin (°F)	RT_{PTS} (°F)	RT_{PTS} (°F) Acceptance Criteria
Lower Shell Longitudinal Weld Seams 3-042 A & B	~~13253~~/34B009+Ni-200, Linde 1092, Lot 3708	~~0.1826~~ ***0.192***	~~0.9825~~ ***1.007***	~~213.2~~ ***221.3***	1.12	~~219.95~~ ***228.31***	-56	65.51	~~229~~ ***238***	270
Lower Shell Longitudinal Weld 3-042C	~~13253~~/34B009+Ni-200, Linde 1092, Lot 3708	~~0.1826~~ ***0.192***	~~0.9825~~ ***1.007***	~~213.2~~ ***221.3***	1.83	~~248.53~~ ***257.98***	-56	65.51	~~258~~ ***267***	270

As seen from the above Table, the 50 EFPY RT_{PTS} value for the Salem Unit 1 Lower Shell Longitudinal Weld 3-042C remains less than the RT_{PTS} Acceptance Criteria of 270 °F for axially-oriented welds as contained in 10 CFR 50.61.

As listed in Salem LRA Table 4.2.4-1, "Summary of the Limiting ART Values used in Generation of the Salem Unit 1 Reactor Vessel Heatup and Cooldown Curves", the weld material representing Lower Shell Longitudinal Weld 3-042C is considered one of the two limiting materials for Salem Unit 1 reactor vessel. Therefore, the changes in wt% Cu and wt% Ni values required re-calculating the Adjusted Reference Temperature (ART) for this weld material. The revised 1/4 T Limiting ART value is 236 °F, which is higher than the ART value of 228 °F originally presented in Salem LRA Table 4.2.4-1.

Due to the revision of the ART value, LRA Table 4.2.4-1 is revised as shown below.

Table 4.2.4-1 Summary of the Limiting ART Values used in Generation of the Salem Unit 1 Reactor Vessel Heatup and Cooldown Curves

EFPY	1/4 T Limiting ART	3/4 T Limiting ART
	Lower Shell Longitudinal Weld Seam 3-042 C	**Intermediate Shell Plate B2402-1**
50	~~228 °F~~ ***236 °F***	177 °F

As discussed in the Disposition to Salem LRA Section 4.2.4 (LRA Page 4-21), Salem Units 1 and 2 will not submit the updates to the P-T and LTOP limit analyses at this time; however, Salem will manage the effects of aging on the intended functions for the period of extended operation. Salem will update the P-T and LTOP limits, based on the latest 1/4 T and 3/4 T Limiting ART values, and submit this information to the NRC to comply with 10 CFR 50 Appendix G. This is a commitment in Salem LRA Appendix A, Section A.5 (No. 49).

Unit 2

Chemistry Factors (CF) Differences for Intermediate Shell Longitudinal Weld Seams 2-442 A, B, & C

Table D-1 of WCAP-15692 shows the Chemistry Factor for the Surveillance Program Weld Material to be 194.53 °F, based on credible surveillance data. The surveillance weld material was fabricated from Heat # 13253. The Salem Unit 2 Intermediate Shell Longitudinal Weld Seams 2-442 A, B, & C were fabricated from Heats #'s 13253 and 20291, Linde 1092, Lot 3833.

Since the surveillance weld material is not identical to the actual material comprising the Intermediate Shell Welds 2-442 A, B, and C, the Salem LRA utilized Position 1.1 of Regulatory Guide 1.99, Rev. 2, to determine the Chemistry Factor as 189.1 °F for the Intermediate Shell Longitudinal Weld Seams 2-442 A, B, and C.

RAI 4.4.2-1

The WCAP-14535-A report, "Topical Report on Reactor Coolant Pump [(RCP)] Flywheel Inspection Examination," is used to support the RCP flywheel analysis for the period of extended operation. To accept this, the Salem plant-specific experience of its RCP flywheels must support all assumptions made in the WCAP-14535-A report analysis. To demonstrate this, please discuss the past examination results of the RCP flywheels, including the associated flaw evaluations conducted.

PSEG Response:

Salem Units 1 and 2 have a combined number of nine (9) Reactor Coolant Pump (RCP) motor flywheels, one for each of the four (4) RCPs per Salem Unit, and a spare.

Salem Units 1 and 2 perform surface and volumetric examinations of all of the RCP motor flywheels in accordance with their respective Technical Specification requirements. Salem Units 1 and 2 RCP motor flywheels have been in operation since August 1976 and April 1980, respectively. As discussed in Table 3-1, "Flywheel Inspection Results" and Table 3-2, "Summary of Recordable Indications", of the WCAP-14535-A report, there has been zero (0) Number of Indications Affecting Flywheel Integrity for Salem Units 1 and 2 (Plant Alpha Designation PSE[Unit 1] / PNJ[Unit 2]) from 1983 through 1995.

In response to this RAI, Salem reviewed ISI flywheel inspection reports both prior to 1983, the period covered by the WCAP-14535-A report, and also from 1995 to present. The review of the flywheel surface and volumetric examinations for the RCP motor flywheels has found that all inspections to date had acceptable results. There were no indications found in any of the ISI inspection reports that required a flaw evaluation to be submitted to the staff for evaluation as required by regulatory position C.4.b(5) of Regulatory Guide 1.14, Rev. 1.

Additionally, in your letter dated September 9, 2005, from Stewart N Bailey, Sr. of the NRC to William Levis of the PSEG Nuclear LLC (ADAMS Accession No. ML051450002), the staff approved Salem Unit 1 Technical Specification Amendment No. 265 and Salem Unit 2 Technical Specification Amendment No. 247 to permit Salem to increase the RCP flywheel inspection to 20 years. This inspection frequency extension was consistent with the Industry/Technical Specification Change Traveler TSTF-421, "Revision to RCP Flywheel Inspection Program (WCAP-15666)" as discussed in PSEG Nuclear LLC letter dated September 27, 2004 to the staff (ADAMS Accession No. ML042790502). The conclusions of WCAP-15666-A state that the results from the WCAP-14535-A report remain valid, and that the extension of the RCP motor flywheel ISI frequency from 10 to 20 years satisfies Regulatory Guide 1.174 criteria as an acceptable change.