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PSEG Nuclear LLC

August 7, 2022

10 CFR 50.90

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LAR S22-01

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

Salem Generating Station Unit 2
Renewed Facility Operating License No. DPR-75
NRC Docket No. 50-311

Subject: **License Amendment Request to Amend the Salem Unit 2 Technical Specifications to Revise and Relocate the Reactor Coolant System Pressure and Temperature Limits and Pressurizer Overpressure Protection System Limits to a Pressure and Temperature Limits Report**

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) is submitting a request for an amendment to the Technical Specifications (TS) for Salem Generating Station Unit 2 (Salem Unit 2).

The proposed amendment would revise the reactor coolant system pressure-temperature (P-T) limits and relocate the pressurizer overpressure protection system (POPS) enable temperature and lift settings and the P-T limits to a Pressure and Temperature Limits Report (PTLR).

Enclosure 1 provides an evaluation supporting the proposed changes. Attachment 1 of Enclosure 1 provides the existing TS pages marked up to show the proposed changes. Attachment 2 of Enclosure 1 provides existing TS Bases pages marked up to show the proposed changes and are being provided for information only. Attachment 3 of Enclosure 1 provides additional justification for application of the WCAP-18124-NP-A methodology for the Salem Unit 2 fluence determination. In addition, WCAP-18124-NP-A, Rev. 0, Supplement 1-NP-A Rev. 0 (ML22153A139) has been approved, which generically approves WCAP-18124-NP-A methodology for use in the extended beltline.

Enclosure 2 provides the Salem Unit 2 Pressure and Temperature Limits Report.

Enclosure 3 contains the non-proprietary Westinghouse analysis (WCAP-16982-NP, Rev. 3) that originally developed the changes to the P-T Limit Curves for license renewal.

Enclosure 4 contains the non-proprietary Westinghouse analysis (WCAP-18571-NP, Rev. 2) that updates the fluence model and validates the changes to the P-T Limit Curves developed for license renewal.

Enclosure 5 provides the non-proprietary version of Westinghouse Letter Report, LTR-SCS-20-52-NP, Revision 1," Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis," (Non-Proprietary Class 3 version with proprietary content redacted) dated July 26, 2021.

Enclosure 6 is an affidavit for withholding Enclosure 7.

Enclosure 7 provides the proprietary version of Westinghouse Letter Report, LTR-SCS-20-52-P, Revision 1, "Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis," dated July 26, 2021. Enclosure 7 contains proprietary information as defined by 10 CFR 2.390. Westinghouse Electric Company LLC (Westinghouse), as the owner of the proprietary information, has executed the affidavit contained in Enclosure 6 pursuant to 10 CFR 20.390(b)(1), and requests that the proprietary information in Enclosure 7 be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4).

PSEG requests approval of this LAR in the normal 12 month review period. The Salem Unit 2 reactor vessel will reach 32 Effective Full Power Years by approximately July 15, 2024. The license amendment will be implemented within 60 days of the issue date.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of New Jersey Official.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Mr. Brian Thomas at 856-339-2022.

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

Executed on August 7, 2022
(Date)

Respectfully,

David Sharbaugh
Site Vice President
Salem Generating Station

Enclosures:
1. Evaluation of Proposed Changes
2. Salem Unit 2 Pressure and Temperature Limits Report (PTLR)
3. Westinghouse Evaluation WCAP-16982-NP, Rev. 3, "Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity," (Non-Proprietary)
4. Westinghouse WCAP-18571-NP, Rev. 2, "Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," (Non-Proprietary)

5. Westinghouse Letter Report LTR-SCS-20-52-NP, Rev. 1, "Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis," (Non-Proprietary)
6. Affidavit for Withholding
7. Westinghouse Letter Report LTR-SCS-20-52-P, Rev. 1, "Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis," (Proprietary)

cc: Administrator, Region I, NRC
Project Manager, NRC
NRC Senior Resident Inspector, Salem
Ms. Ann Pfaff, Manager, NJBNE
PSEG Corporate Commitment Tracking Coordinator
Site Commitment Tracking Coordinator

Enclosure 1

Evaluation of Proposed Changes

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1.0 SUMMARY DESCRIPTION

The proposed change revises the Salem Unit 2 Technical Specifications (TS) reactor coolant system pressure-temperature (P-T) limits and relocates the pressurizer overpressure protection system (POPS) enable temperature and lift settings and the P-T limits to a Pressure and Temperature Limits Report (PTLR).

The PTLR contains updates to the P-T limit curves for the beltline regions for the Salem Unit 2 reactor pressure vessel (RPV). The P-T limit curves are developed for 50 effective full power years (EFPY) of operation. The pressure and temperature limits were developed using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G (Reference 1). The methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (Reference 2) and WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," (Reference 8).

The proposed TS changes are consistent with the guidance of NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," (Reference 3), and with Technical Specification Task Force (TSTF) Travelers TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," (Reference 4), as clarified in the NRC's August 4, 2011 letter (Reference 7) and TSTF-233, "Relocate LTOP Arming Temperature to PTLR," (Reference 6).

2.0 DETAILED DESCRIPTION

2.1 <u>System Design and Operation</u>

Pressure-temperature limit curves have been developed for the RPV in accordance with 10 CFR 50 Appendix G, ASME Code Section XI, Appendix G, and NRC Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*, as applicable. These P-T curves account for material property changes due to radiation and ensure that a postulated surface defect having a depth of 1/4 of the RPV material thickness can be safely accommodated in the vessel shell without promoting brittle fracture.

2.2 <u>Current Technical Specifications Requirements</u>

TS Limiting Condition for Operation (LCO) 3.4.10.1 requires that Reactor Coolant System (RCS) pressure and temperature be maintained within the pressure and temperature limits specified in TS Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality and inservice leak and hydrostatic testing. The current P-T limit curves in the TSs are applicable to plant operation up to 32 Effective Full Power Years (EFPY). LCO 3.4.10.1 also requires the temperature rate of change to be maintained within specified limits during heatup, cooldown and hydrostatic testing operations above system design pressure.

TS 3.4.10.3 requires the lift setting for the POPS relief valves to be maintained less than or equal to the specified limit when the temperature of one or more of the RCS cold legs is less than or equal to the specified POPS enable temperature.

The POPS enable temperature is also specified in TS 3.1.2.3, 3.4.1.3, 3.4.1.4 and 3.5.3.

2.3 Reason for the Proposed Change

The current Unit 2 P-T limit curves expire at 32 EFPY, which Salem Unit 2 is expected to reach by approximately July 15, 2024. The requested approval date for the proposed amendment supports establishment of updated P-T limit curves prior to reaching 32 EFPY.

2.4 Description of the Technical Specification Changes

This license amendment request revises the reactor coolant system P-T limit curves (i.e., TS Figures 3.4-2 and 3.4-3) and relocates the POPS enable temperature and lift settings and the P-T limits to a PTLR, as follows:

1. Revise Index to add new PTLR definition 1.20a and update page number for Section 6.9.2.
2. Add a definition in Section 1.0, 1.20a, for the PTLR.
3. Revise TS 3.1.2.3, Charging Pump – Shutdown, “#” note to change the RCS cold leg temperature from the current value of 312°F to “the POPS enable temperature specified in the PTLR.”
4. Revise TS 3.4.1.3, “*” note to change the RCS cold leg temperature from the current value of 312°F to “the POPS enable temperature specified in the PTLR.”
5. Revise TS 3.4.1.4, “##” note to change the RCS cold leg temperature from the current value of 312°F to “the POPS enable temperature specified in the PTLR.”
6. Revise TS 3.4.10.1 Limiting Condition for Operation, to refer to limits specified in the PTLR.
7. Revise TS 3.4.10.1.a, the maximum heatup rate, to refer to limits specified in the PTLR.
8. Revise TS 3.4.10.1.b, the maximum cooldown rate, to refer to limits specified in the PTLR.
9. Revise TS 3.4.10.1.c, the maximum temperature change, to refer to limits specified in the PTLR.
10. Revise Surveillance Requirements (SR) 4.4.10.1.1 to refer to the limits specified in the PTLR.
11. Revise SR 4.4.10.1.2 to refer to the Figures (P-T Limit Curves) specified in the PTLR.
12. Remove TS Figure 3.4-2, “Salem Unit 2 Reactor Coolant System Heatup Limitations.”

13. Remove TS Figure 3.4-3, “Salem Unit 2 Reactor Coolant System Cooldown Limitations.”

14. Revise TS 3.4.10.3 to refer to the POPS enable temperature and relief valve setting specified in the PTLR.

15. Revise TS 3.5.3, “#” note to change the RCS cold leg temperature from the current value of 312°F to “the POPS enable temperature specified in the PTLR.”

16. Revise SR 4.5.3.2 to change the RCS cold leg temperature value of 312°F, to referto the POPS enable temperature specified in the PTLR.

17. Adds a new Specification 6.9.1.11, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” to subsection 6.9, “Reporting Requirements” in Section 6.0, “Administrative Controls.” The new specification is consistent in format and content with the NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Revision 5.0 and includes:

    - The individual TSs that address reactor coolant system P-T limits,
    - References the NRC approved topical report which documents the PTLR methodology, and
    - Requires the PTLR and any revisions or supplements to be submitted to the NRC.

The marked-up TS pages are provided in Attachment 1.

2.5 Description of Technical Specification Bases Changes

The proposed changes to the TS Bases provided in Attachment 2 are provided for information only; changes to the affected TS Bases pages will be incorporated in accordance with TS 6.16, “Technical Specifications (TS) Bases Control Program.”

3.0 TECHNICAL EVALUATION

3.1 Background

In Salem Unit 2 Amendment 224 (Reference 5), the curves were calculated using the Westinghouse vessel fluence methodology. Allowable pressure-temperature relationships for various heatup and cooldown rates were calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-14040-NP-A, Rev. 2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," approved March 1999. Adjusted reference temperatures at the nil ductility transition values were developed for the RPV materials in accordance with RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials.” The NRC staff approved revising the RPV pressure-temperature limits and extending their validity to 32 effective full-power years. These are the current P-T limits in the Salem Unit 2 TS.

3.2 Technical Analysis

Generic Letter 96-03 provides regulatory guidance regarding relocation of P-T curves and associated numerical limits (such as heatup and cooldown rates) from plant TS to a PTLR (a licensee controlled document). As stated in GL 96-03, a licensee requesting such a change must satisfy the following three criteria:

1. Have NRC-approved methodologies to reference in the TS.

2. Develop a PTLR to contain the P-T limit curves, associated numerical limits, and any necessary explanation, and

3. Modify applicable sections of the TS accordingly.

This License Amendment Request is based on methodologies described in WCAP-14040-A, Rev. 4 with WCAP-18124-NP-A, Rev 0 used as an alternative to WCAP-14040-A, Rev. 4, Section 2.2.

The following table lists the provisions specified for an approved methodology and the information required to be included in the PTLR, as shown in Attachment 1 of GL 96-03 and the corresponding information in the Salem Unit 2 PTLR.

	PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	PSEG PTLR (Enclosure 2) COMPLIANCE
1.	The methodology shall describe how the neutron fluence is calculated (reference new regulatory guide when it is issued).	Provide the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.	Table 4-1 provides the values of the neutron fluences used in the ART calculation.
2.	The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.	Provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located. Reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data.	Appendix A provides the surveillance capsule withdrawal schedule. Appendix A also contains the Westinghouse surveillance capsule reports of the four capsules withdrawn to date.
3.	Low temperature overpressure protection (LTOP) system limits developed using NRC- approved methodologies may be included in the PTLR.	Provide setpoint curves or setpoint values	Section 4.14 provides the LTOP system setpoint value.
4.	The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.	Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (t = vessel beltline thickness). PWRs - identify RT_{PTS} value in accordance with 10 CFR 50.61.	Table 4-6 provides the limiting ART values and the limiting materials. Section 4.11 references the location of the RT_{PTS} values in the Westinghouse analysis (Enclosure 4).
5.	The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG- 0800, SRP Section 5.3.2, Pressure-Temperature Limits.	Provide the P/T curves for heatup, cooldown, criticality, and hydrostatic and leak tests.	Figures 3-1 and 3-2 provide the P/T curves, including heatup, cooldown, criticality, and hydrostatic and leak tests.

	PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	PSEG PTLR (Enclosure 2) COMPLIANCE
6.	The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.	Identify minimum temperatures on the P/T curves such as minimum boltup temperature and hydrotest temperature.	Figures 3-1 and 3-2 P/T curve annotations provide the minimum boltup and hydrotest (hydrostatic) temperatures.
7.	Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2σ) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation. Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.	Sections 4.5 and 4.6 provide descriptions of how the data from multiple surveillance capsules are used in the chemistry factor and ART calculation. Appendix C contains the credibility evaluation of the surveillance data.

Revised pressure-temperature curves were developed for hydrostatic pressure and leak tests, core not critical, and core critical conditions. The PTLR containing the P-T limit curves, associated tabulated numerical limits, and necessary explanation, is provided in Enclosure 2. The revised curves have been developed for 50 EFPY.

The P-T limit curves were generated through End of License Extension (EOLE), i.e., 60 years of operation, using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G. The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4.

The NRC concluded WCAP-14040-A, Revision 4 was acceptable for referencing as a PTLR methodology, subject to three conditions:

a. *Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.*

   The methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 require that licensees briefly describe the surveillance program. Licensees should identify by title and number the report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. The RPV surveillance program was verified to comply with the Requirements of 10 CFR 50, Appendix H in Section 3.0.3.2.9, Reactor Vessel Surveillance, of the Salem Units 1 and 2 license renewal safety evaluation report (SER) (Reference 11). The requirement to adhere to 10 CFR 50, Appendix H is retained in TS SR 4.4.10.1.2. The PTLR specifically identifies:

   - The surveillance program's compliance with 10 CFR 50, Appendix H,
   - That the surveillance capsule withdrawal schedule is contained in UFSAR Section 5.2.4.4, and
   - A complete listing of the Reactor Vessel Surveillance Program and surveillance capsule reports

b. *Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.*

   This condition was satisfied upon publication of WCAP-14040-A, Revision 4.

c. *As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.*

The Salem Unit 2 P-T limit curves were developed with the flange requirements of 10 CFR 50, Appendix G.

The Westinghouse evaluation supporting the P-T Limit Curves, WCAP-18571-NP, Rev. 2, “Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation,” (Reference 9) is contained in this submittal as Enclosure 3. This report verifies the P-T Limit Curves that were generated in Westinghouse evaluation WCAP-16982-NP, Rev. 3 for 50 EFPY (Reference 10) developed for license renewal remain applicable for Salem Unit 2. Refer to Enclosure 4 for this report. The reports’ contents are summarized below.

Section	WCAP-16982-NP, Rev. 3	WCAP-18571-NP, Rev. 2
1.0	Identifies the RPV analyses which are considered Time-Limited Aging Analyses for License Renewal.	Describes the evaluation to confirm the applicability of the P-T curves from WCAP-16982-NP with consideration of updated fluence values.
2.0	References the methodology for the fluence projections (use of DORT discrete ordinates code Version 3.2 and the BUGLE-96 cross-section library), as well as the results of the projections.	Provides the methodology for the fluence projections (use of 3-dimensional RAPTOR-M3G), as well as the results of the projections, including fluence at the primary nozzle forgings.
3.0	Evaluates the fracture toughness properties of the RPV beltline and extended beltline materials, including a summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Salem Unit 2 RPV materials.	Evaluates the fracture toughness properties of the RPV beltline and extended beltline materials, including a summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Salem Unit 2 RPV materials (including nozzle forgings).
4.0	Summarizes the Pressurized Thermal Shock (PTS) results.	Discusses the results of the Salem Unit 2 surveillance capsule program, as well as Salem Unit 2 applicable sister plant surveillance capsule data.
5.0	Provides the Upper Shelf Energy (USE) values at 50 EFPY with a comparison to the screening criteria.	Describes the calculation of Chemistry Factors using Regulatory Guide 1.99, Revision 2 for the Salem Unit 2 RPV materials and its sister plant's materials.

Section	WCAP-16982-NP, Rev. 3	WCAP-18571-NP, Rev. 2
6.0	Explains the detailed methodology for developing the Salem Unit 2 P-T Limit Curves, including requirements to address the metal temperature of the closure head/vessel flange and the minimum boltup temperature. Also discusses the minimum LTOP enable temperature. Provides the tabulated results supporting the Salem Unit 2 P-T Limit Curves, as well as the generated curves.	Explains the Adjusted Reference Temperature (ART) calculation methodology and results at the 1/4T and 3/4T RPV locations through 50 EFPY, including the determination of the limiting ART values
7.0	Provides the surveillance capsule withdrawal schedule, including the recommendation for the next capsule withdrawal.	Provides the basis for P-T Curve applicability for those P-T Curves developed in WCAP-16982-NP, Rev. 3 (refer to Section 6.0 of that report).
8.0	Determines the emergency response guideline (ERG) P-T Limit category.	Lists References for the report.
9.0	Lists References for the report.	Not Applicable to this Report.
Appendix A	Provides Stress Intensity Factors (K_{IT}) at 1/4T and 3/4T locations for 50 EFPY.	Describes the evaluation of the credibility of the Salem Unit 2 surveillance capsule program data using the criteria provided in Regulatory Guide 1.99, Revision 2.
Appendix B	Not Applicable to this Report.	Contains the Pressurized Thermal Shock (PTS) evaluation as required by 10 CFR 50.61 and ERG classification.
Appendix C	Not Applicable to this Report.	Validation of the radiation transport models based on neutron dosimetry measurements

<u>Neutron Fluence Calculations:</u>

The methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET." The neutron transport evaluation methodology described in the WCAP report is based on the guidance of Regulatory Guide 1.190. The NRC concluded the calculational methodology described in WCAP-18124-NP-A, Revision 0 is acceptable for use in calculating RPV neutron fluence provided the following limitations and conditions are met:

1. *Applicability of WCAP-18124-NP, Revision 0, is limited to the reactor pressure vessel region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to response parameters of interest (for example, pressure-temperature limits, material stress and strain), margin*

assessment, or a combination thereof, for applications of the method to components including, but not limited to, the reactor pressure vessel upper circumferential weld, and reactor coolant system inlet and outlet nozzles and reactor vessel internal components.

The methodology described in WCAP-18124-NP-A, Revision 0 was applied to Salem Unit 2 extended beltline materials and RCS inlet and outlet nozzles. Additional justification for the Salem Unit 2 application is provided in Attachment 3 to this enclosure (Reference 13). Based on the fluence uncertainty analysis, benchmarking, and margin assessment described in Attachment 3, the applicability of the RAPTOR-M3G fluence determination methodology is justified for the Salem Unit 2 RPV extended beltline region and RCS inlet and outlet nozzle fluence determination. In addition, WCAP-18124-NP-A, Rev. 0, Supplement 1-NP-A Rev. 0 (ML22153A139) has been approved, which generically approves WCAP-18124-NP-A methodology for use in the extended beltline.

2. *Least squares adjustment is acceptable if the adjustments to the measured or calculated ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the inconsistency should be disqualified.*

   This limitation applies in situations where the least squares analysis is used to adjust the calculated values of neutron exposure. For Salem Unit 2, Limitation and Condition 2 does not apply as the least-squares procedures were not used to adjust the calculated fast neutron (E > 1.0 MeV) fluence values for RPV materials evaluated in the updated reactor vessel integrity analysis (Enclosure 4). The least-squares results were only used to compare the calculations and measurements from the evaluated dosimetry and validate the neutron transport models, and those comparisons showed satisfactory results.

A validation of the fluence model is provided in Appendix C of Enclosure 4.

The fluence is based upon operation for 50 EFPY. Cycle-specific calculations were performed for Cycles 1 through 24. Note that future fluence projection data beyond Cycle 24 are based on Cycles 20 and 23.

The calculated fast neutron fluences at the end of plant life (50 EFPY) are provided below:

Parameter	Fluence (n/cm^2)
Peak Surface (Intermediate Shell Plates)	$2.05x10^{19}$
Peak ¼ T (Intermediate Shell Plates)	$1.22x10^{19}$
Limiting Beltline Material Peak Surface (Intermediate Shell Longitudinal Weld Seams 2-442 B & C)	$1.47x10^{19}$
Limiting Beltline Material Peak ¼ T (Intermediate Shell Longitudinal Seams 2-442 B & C)	$0.876x10^{19}$

<u>Use of Salem Unit 2 and Sister Plant Surveillance Capsule Results and Adjusted Reference Temperature:</u>

As discussed in Appendix C of the PTLR, the surveillance data for intermediate shell plate B4712-2 (Salem Unit 2 surveillance program) and lower shell longitudinal weld seams 3-442A, B, and C (sister plant Diablo Canyon Unit 2 surveillance program) are deemed credible.

As reported in WCAP-18571-NP, Rev. 2, the Diablo Canyon Unit 2 surveillance data for Heat # 21935/12008 were deemed credible, hence the calculated lower Adjusted Reference Temperature (ART) of Salem Unit 2 lower shell longitudinal weld seams 3-442A, B, and C calculated as determined by Regulatory Guide 1.99, Rev. 2 Position 2.1 supersedes those calculated with Regulatory Guide 1.99, Rev, 2, Position 1.1. Therefore, the limiting material for Salem Unit 2 are intermediate shell longitudinal weld seams 2-442B and C.

This Diablo Canyon Unit 2 surveillance program is applicable to Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442 A, B, and C since both the surveillance weld and reactor vessel welds were fabricated with wire Heat # 21935/12008 with Linde 1092 flux. The surveillance data is reported in WCAP-17315-NP, “Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations,” and were adjusted to account for difference in weld chemistry and irradiation temperature. This approach is similar to that taken for the McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate, evaluated in WCAP-17455-NP, “McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations” and approved in the SER (ML13073A041), which also used Diablo Canyon Unit 2 surveillance data to evaluate a McGuire Unit 1 reactor vessel weld with wire Heat # 21935/12008.

Applicability of WCAP-16982-NP P-T Limit Curves:

Instead of re-creating the P-T curves in WCAP-18571-NP, Rev. 2, the P-T curves in WCAP-16982-NP, Rev. 3 were evaluated for applicability for 50 EFPY. Below is a summary of the Adjusted Reference Temperatures (ART) needed for this applicability evaluation (re-produced from Table 7-1 of WCAP-18571-NP, Rev. 2).

Summary of the Limiting ART Values

1/4T Limiting ART (°F)		**3/4T Limiting ART (°F)**	
WCAP-16982-NP, Rev. 3	WCAP-18571-NP, Rev. 2	WCAP-16982-NP, Rev. 3	WCAP-18571-NP, Rev. 2
209	191.6	150	138.1

The above table shows that the limiting ART values at the 1/4T and 3/4T locations will not exceed the ART values used in the P-T limit curves of WCAP-16982-NP. Furthermore, the limiting initial RT_{NDT} value of 15°F for the closure head flange and the vessel flange remains appropriate, and the P-T limit curves flange notch and bolt-up temperature require no change or further consideration. Therefore, the EOLE (50 EFPY) P-T limit curves generated in WCAP-16982-NP can be implemented in the PTLR.

Pressure-Temperature Curve Evaluation:

The beltline region of the RPV was evaluated to develop the 50 EFPY P-T curves (Reference 10), which were re-validated in Reference 9. The WCAP-104040-A, Rev. 4 methodology used to generate the P-T curves in this submittal is approved by the NRC. When re-validating the P-T Curves in Reference 9, the credible surveillance capsule data were incorporated into the ART values in accordance with RG 1.99, Rev. 2.

The 50 EFPY P-T curves ensure that adequate RPV safety margins against non-ductile failure will continue to be maintained during normal operations, anticipated operational occurrences, and inservice leak and hydrostatic testing. Together, these measures ensure that the integrity of the Reactor Coolant System will be maintained for the life of the plant.

Proposed revisions to applicable sections of the TS have been prepared and are provided in Attachment 2 to this submittal. These proposed changes are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419 and TSTF-233.

Pressurizer Overpressure Protection System (POPS) Power Operated Relief Valves (PORVs) setpoint selection

The Low Temperature Overpressure Protection System (LTOPS), known as the Pressurizer Overpressure Protection System (POPS) at Salem, provides RCS pressure relief capability to mitigate the overpressure transients that may occur during cold shutdown, heatup, and cooldown operations to minimize the potential for challenging reactor vessel integrity limits when operating at low temperature conditions (i.e., 10 CFR 50, Appendix G limits). At Salem, the pressurizer Power Operated Relief Valves (PORVs), with reduced lift settings, provide a method of overpressure protection. The POPS PORV setpoints are selected in accordance with the NRC approved methodology in WCAP-14040-A, Rev. 4, such that the peak pressure during the design basis Mass Injection (MI) and Heat Injection (HI) transients at low RCS temperature conditions (i.e., ≤ 350°F) will not exceed the isothermal Appendix G Pressure-Temperature (P-T) limits. Updated P-T limits have been developed for Salem Unit 2 through a service life of 50 EFPY. As part of updating the P-T limits, an analysis was performed (Reference 12) to validate the applicability of the POPS analysis previously performed (Reference 14) to ensure that the POPS configuration, relief valve setpoints, and operating limitations protect the revised P-T limits at 50 EFPY.

The following acceptance criteria were used to determine the POPS PORV setpoints:

1. The peak RCS pressure resulting from the design basis MI and HI transients shall not exceed the minimum of the steady-state adjusted Appendix G limits and the PORV piping limit.

2. The minimum RCS pressure resulting from the design basis MI and HI transients should not drop below the RCP No. 1 Seal ΔP limit.

The maximum allowable PORV setpoint is determined based on the adjusted Appendix G limit or the PORV piping limit, whichever is more limiting. The adjusted Appendix G limit is the Appendix G limit minus the transmitter ΔP and wide range pressure instrument uncertainty. The maximum allowable PORV setpoints for the MI and HI transients were determined as a function of indicated RCS temperature, where the final maximum allowable PORV setpoint would bound both the MI and HI transient maximum allowable PORV setpoints.

The maximum allowable Salem Unit 2 POPS PORV setting was determined by the analysis to be 434 psig. The current Salem Unit 2 POPS PORV setting of 375 psig will be documented in the PTLR to maintain conservative margin.

The POPS enable temperature of 292°F with temperature uncertainty applied was determined by Westinghouse Letter Report LTR-SCS-20-52 (Reference 12) using ASME code case N-641.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in the TSs which includes limiting conditions for operation (LCO's), surveillance requirements and administrative controls. Previously the plant-specific P-T limits had been

incorporated into the TS and controls were placed on operation and testing by the associated specification. This proposed change revises the TS to relocate the P-T limit curves to a licensee controlled document in accordance with the guidance of Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," and TSTF-233, "Relocate LTOP Arming Temperature to PTLR."

10 CFR 50.60 requires that light-water nuclear power reactors meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in Appendix G to 10 CFR 50. Appendix G is the regulatory basis for P-T curves for light water reactors. Appendix G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Appendix G also requires that the reference temperature and Charpy upper-shelf energy for reactor vessel beltline materials account for the embrittlement caused by neutron fluence over the life of the vessel.

The NRC has established requirements in 10 CFR 50, Appendix G, "Fracture Toughness Requirements," in order to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. Appendix G requires that the P-T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods and margins of safety of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code were used to generate the P-T limits. 10 CFR Part 50, Appendix G contains additional requirements for the metal temperature of the closure head flange and vessel flange regions or when the core is critical. Also, Appendix G requires that applicable surveillance data from RPV material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

Appendix H to 10 CFR Part 50 provides requirements related to facility RPV material surveillance programs. Salem Unit 2 demonstrates its compliance with the requirements of 10 CFR 50, Appendix H through withdrawal and analysis of surveillance capsules in accordance with the schedule contained in the UFSAR.

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

4.2 <u>Precedents</u>

The NRC has approved similar license amendments to relocate the P-T limit curves to a PTLR. Examples include:

- Salem Generating Station, Unit 1, (License Amendment No. 339 issued by NRC letter dated October 8, 2021, ADAMS Accession No. ML21230A018)
- Hope Creek Generating Station, (License Amendment No. 209 issued by NRC letter dated December 14, 2017, ADAMS Accession No. ML17324A840)
- Palo Verde Nuclear Generating Station (License Amendment No. 178 issued by NRC letter dated February 25, 2010, ADAMS Accession No. ML100480188)
- Sequoyah Nuclear Plant Units 1 and 2, (License Amendment Nos. 294/284 issued by NRC

letter dated September 15, 2004, ADAMS Accession No. ML042600465)

- Prairie Island Nuclear Generating Plant Units 1 and 2, (License Amendment Nos. 135/127 issued by NRC letter dated May 4, 1998, ADAMS Accession No. ML022260558)
- Beaver Valley Power Station Units 1 and 2, (License Amendment Nos. 256/138 issued by NRC letter dated July 15, 2003, Accession No. ML031960399)
- Byron/Braidwood, (License Amendment Nos. 98 (Byron) and 89 (Braidwood) issued by NRC letter dated January 23, 1998, ADAMS Accession No. ML020860605)

The NRC previously approved changes similar to the proposed pressure and temperature limit change in this License Amendment Request for other nuclear power plants. Examples include:

- Arkansas Nuclear One Unit 2, (License Amendment 311 issued by NRC letter dated November 27, 2018, ADAMS Accession No. ML18298A012)
- Catawba Nuclear Station Units 1 and 2, (License Amendment Nos. 306/302 issued by NRC letter dated August 4, 2020, ADAMS Accession No. ML20174A045)
- R. E. Ginna Nuclear Power Plant, (License Amendment No. 106 issued by NRC letter dated February 23, 2009, ADAMS Accession No. ML083530806)
- H. B. Robinson Steam Electric Plant Unit No. 2, (License Amendment No. 248 issued by NRC letter dated November 22, 2016, ADAMS Accession No. ML16285A404)
- Indian Point Generating Unit No. 3, (License Amendment No. 258 issued by NRC letter dated September 3, 2015, ADAMS Accession No. ML15226A159)
- Seabrook Station Unit No. 1, (License Amendment No. 151 issued by NRC letter dated November 2, 2015, ADAMS Accession No. ML15096A255)
- Joseph M. Farley Nuclear Plant Units 1 and 2, (Amendment Nos.193/189 issued by NRC letter dated October 2, 2013, ADAMS Accession No. ML13249A386)

4.3 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) requests an amendment to Renewed Facility Operating License No. DPR-75 for Salem Generating Station Unit 2 (Salem Unit 2). The proposed Technical Specification (TS) changes modify the Salem Unit 2 TS by revising the pressure and temperature (P-T) limits and relocate the pressurizer overpressure protection system (POPS) enable temperature and lift settings and the P-T limits to a Pressure and Temperature Limits Report (PTLR).

PSEG has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change revises the pressure and temperature (P-T) limits and relocates the pressurizer overpressure protection system (POPS) enable temperature and lift settings and the P-T limits to a Pressure and Temperature Limits Report (PTLR). The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4 with WCAP-18124-NP-A used as an alternative to WCAP-14040-A, Revision 4, Section 2.2.

   10 CFR 50 Appendix G establishes requirements to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants.

   Implementing this NRC approved methodology does not reduce the ability to protect the RCPB as specified in Appendix G, nor will this change increase the probability of malfunction of plant equipment, or the failure of plant structures, systems, or components. Incorporation of the new methodology for calculating P-T curves, and the relocation of the P-T curves, POPS Enable Temperature, and POPS limits from the TS to the PTLR provides an equivalent level of assurance that the RCPB is capable of performing its intended safety functions.

   The proposed changes do not adversely affect accident initiators or precursors, and do not alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety functions is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The change in methodology for calculating P-T limits and the relocation of those limits to the PTLR do not alter or involve any design basis accident initiators. RCPB integrity will continue to be maintained in accordance with 10 CFR 50 Appendix G, and the assumed accident performance of plant structures, systems and components will not be affected. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), and the installed equipment is not being operated in a new or different manner.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed changes involve a significant reduction in a margin of safety?

   Response: No

The proposed changes do not affect the function of the RCPB or its response during plant transients. Calculating the Salem Unit 2 P-T limits using the NRC approved methodology ensures adequate margins of safety relating to RCPB integrity are maintained. The proposed changes do not alter the manner in which the Limiting Conditions for Operation P- T limits for the RCPB are determined. The operability requirements for equipment assumed to operate for accident mitigation are not affected.

Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, PSEG concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

Therefore, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. 1998 through the 2000 Addenda Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G

2. WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” dated May, 2004 (ADAMS Accession No. ML050120209)

3. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996 (ADAMS Accession No. 9601290350)

4. TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," dated August 4, 2003 (ADAMS Accession No. ML012690234)

5. Salem Nuclear Generating Station, Unit Nos. 1 and 2, Issuance of Amendment Re: Increase Licensed Power Levels From 3,411 MWT to 3,459 MWT, dated May 25, 2001

(ADAMS Accession No. ML011350051)

6. TSTF-233, “Relocate LTOP Arming Temperature to PTLR,” dated July 16, 1998

7. NRC Letter, “Implementation of Travelers TSTF-363, Revision 0, “Revise Topical Report References In ITS 5.6.5, COLR [Core Operating Limits Report],” TSTF-408, Revision 1, “Relocation of LTOP [Low Temperature Overpressure Protection] Enable Temperature and PORV [Power-Operated Relief Valve] Lift Setting To the PTLR [Pressure-Temperature Limits Report],” and TSTF-419, Revision 0, “Revise PTLR Definition and References In ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR,” dated August 4, 2011 (ADAMS Accession No. ML110660285)

8. WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET," dated July, 2018 (ADAMS Accession No. ML18204A010)

9. WCAP-18571-NP, Revision 2, “Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation,” dated July 2022

10. WCAP-16982-NP, Revision 3, “Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity,” dated December 2020

11. Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station, Units 1 and 2, June 13, 2011 (ADAMS Accession No. ML11164A051)

12. Westinghouse Letter Report LTR-SCS-20-52-P, Rev. 1, “Salem Unit 2 Low Temperature Overpressure Protection System (LTOPS) / Pressurizer Overpressure Protection System (POPS) Analysis dated July 26, 2021

13. Westinghouse Letter PNJ-REAC-TM-AA-000001, Rev. 1, dated July 6, 2022, “Justification of Using RAPTOR-M3G for Reactor Pressure Vessel Extended Beltline Materials at Salem Unit 2”

14. Westinghouse Letter PSE-09-18, Revision 0, “Transmittal of LTOPS Setpoint Evaluation for Salem Units 1 and 2”, March 2009

Attachment 1

Mark-up of Proposed Technical Specification Changes

The following Technical Specification pages for Renewed Facility Operating License DPR-75 are affected by this change request:

I
XVIII
1-5
3/4 1-9
3/4 4-3
3/4 4-4a
3/4 4-27
3/4 4-28
3/4 4-29
3/4 4-31
3/4 5-7
3/4 5-8
6-24b
6-24c (new page)

INSERT 1 (Page 1-5)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.20a The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the Overpressure Protection System setpoint and enable temperature, for the current reactor vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Technical Specification Section 6.9.1.11.

INSERT 2 (Page 6-24b)

6.9.1.11 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, POPS enable temperature, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.1.2.3, "Charging Pump-Shutdown"
2. Specification 3.4.1.3, “Reactor Coolant System Hot Shutdown”
3. Specification 3.4.1.4, “Reactor Coolant System Cold Shutdown"
4. Specification 3/4.4.10.1, "Reactor Coolant System Pressure/Temperature Limits"
5. Specification 3.4.10.3, "Reactor Coolant System Overpressure Protection Systems"
6. Specification 3/4.5.3, “Emergency Core Cooling Systems ECCS Subsystems - Tavg < 350°F”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC.

1. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May, 2004.

2. WCAP-18124-NP-A, Rev 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018, may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

SECTION PAGE

1.0 DEFINITIONS

INDEX

ADMINISTRATIVE CONTROLS

===

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

Insert 1

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.#

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

the POPS enable temperature specified in the PTLR,

\# A maximum of one centrifugal charging pump shall be OPERABLE while in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to ~~312°F~~, MODE 5, or MODE 6 when the head is on the reactor vessel.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 21 and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop 22 and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop 23 and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop 24 and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal Loop 21,

6. Residual Heat Removal Loop 22.

b. At least one of the above coolant loops shall be in operation.**

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

the POPS enable temperature specified in the PTLR

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to ~~312°F~~ unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 92% of level) or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

**All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.##

ACTION:

a. With less than the above required loops operable, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation; suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

One RHR loop may be inoperable for up to two hours for surveillance testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop.

the POPS enable temperature specified in the PTLR

A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to ~~312°F~~ unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 92% of level), or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

* Systems supporting RHR loop operability may be excepted as follows:

  a. The normal or emergency power source may be inoperable.

** The residual heat removal pumps may be de-energized for up to 2 hours provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

limits specified in the PTLR with:

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the ~~limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with~~:

a. A maximum heatup ~~of 100°F in any one hour period,~~

rate within the limits specified in the PTLR,

b. A maximum cooldown ~~of 100°F in any one hour period,~~ and

c. A maximum temperature change ~~of less than or equal to 5°F in any one hour period,~~ during hydrostatic testing operations above system design pressure.

within the limits specified in the PTLR

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

specified in the PTLR

4.4.10.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits ~~in~~ accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.10.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update ~~Figures 3.4-2 and 3.4-3.~~

the P-T Limit Curves specified in the PTLR.

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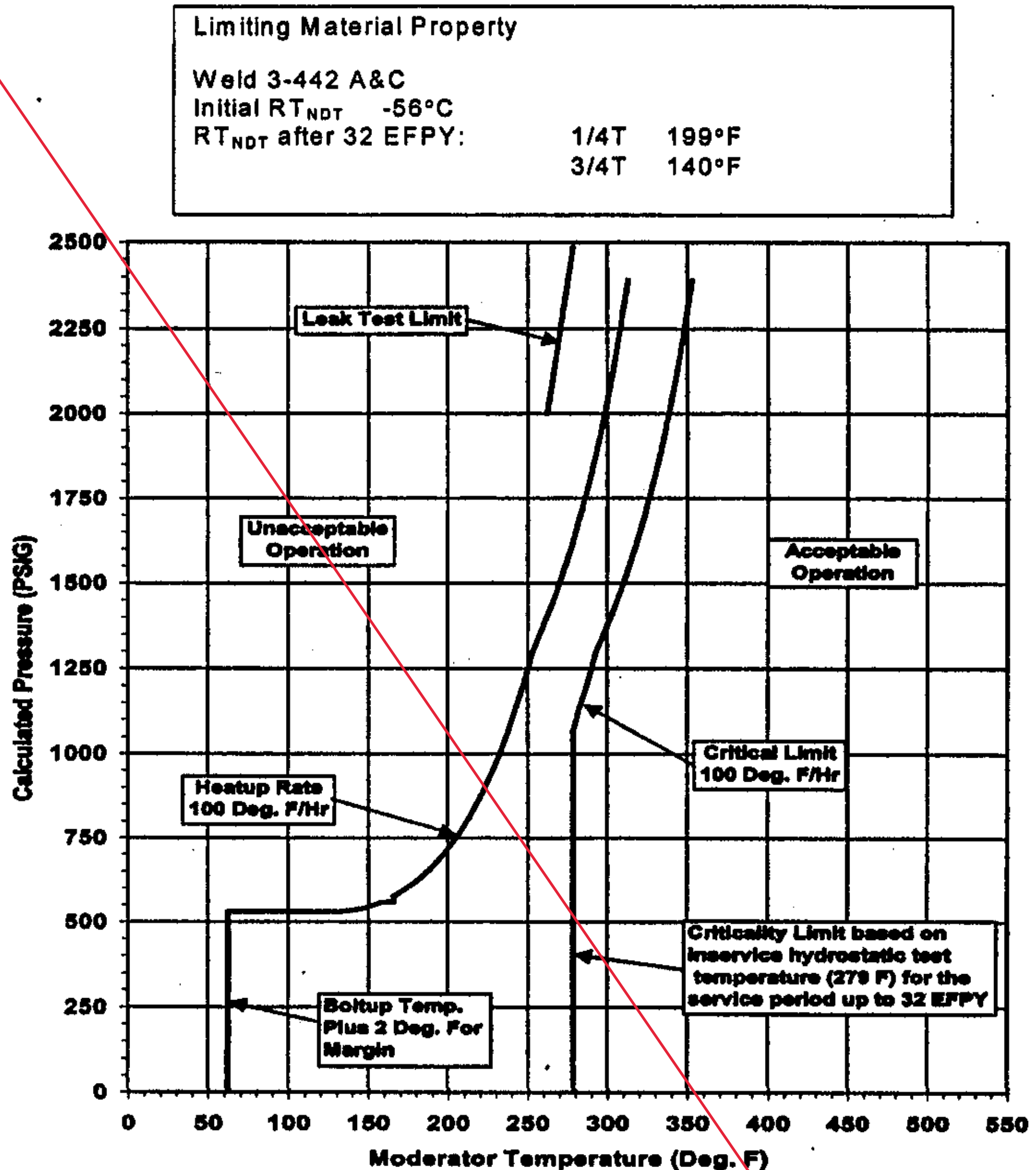

Figure 3.4-2

SALEM UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 32 EFPY WITH MAXIMUM HEATUP RATE OF 100 F/HR. CURVE CONTAINS MARGIN FOR INSTRUMENT ERRORS

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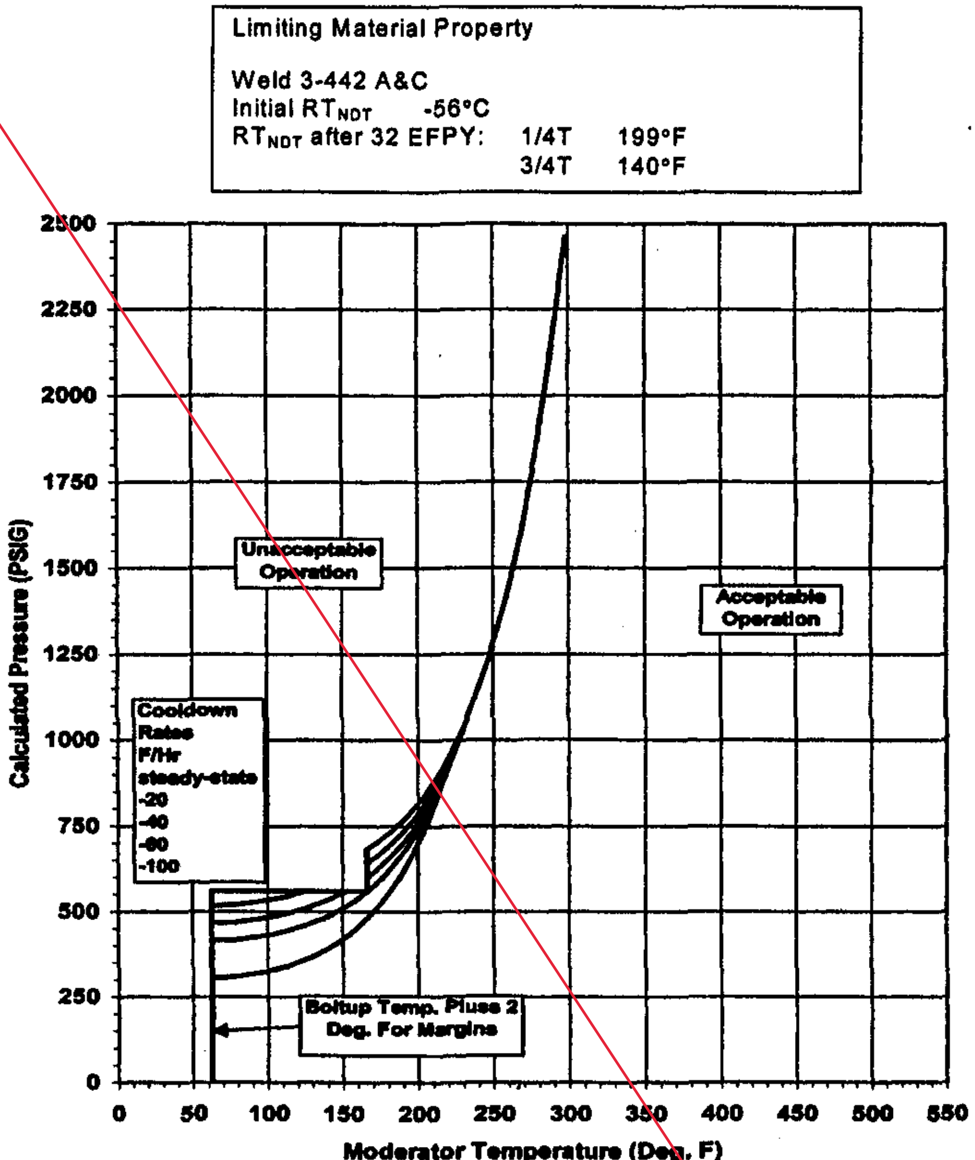

Figure 3.4-3

SALEM UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 32 EFPY WITH MAXIMUM COOLDOWN RATE OF 100 F/HR. CURVE CONTAINS MARGIN FOR INSTRUMENT ERRORS.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two Pressurizer Overpressure Protection System relief valves (POPS) with a lift setting of less than or equal to ~~375 psig~~, or

the value specified in the PTLR,

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to ~~312°F~~, except when the reactor vessel head is removed.

the POPS enable temperature specified in the PTLR,

ACTION:

a. With one POPS inoperable in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to ~~312°F~~, restore the inoperable POPS to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

b. With one POPS inoperable in MODES 5 or 6 with the Reactor Vessel Head installed, restore the inoperable POPS to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

c. With both POPSs inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

d. In the event either the POPS or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the POPS or vent(s) on the transient and any corrective action necessary to prevent recurrence.

e. LCO 3.0.4.b is not applicable when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.10.3.1 Each POPS shall be demonstrated OPERABLE by:

ECCS SUBSYSTEMS - Tavg <350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump # and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

1. Discharging into each Reactor Coolant System (RCS) cold leg.

b. One OPERABLE residual heat removal pump and associated residual heat removal heat exchanger and flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation and;

1. Discharging into each RCS cold leg, and; upon manual initiation,

2. Discharging into two RCS hot legs.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} lessthan 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

d. LCO 3.0.4.b is not applicable to ECCS high head subsystem

A maximum of one safety injection pump or one centrifugal charging pump shall be OPERABLE in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to ~~312°F,~~ Mode 5, or Mode 6 when the head is on the reactor vessel.

the POPS enable temperature specified in the PTLR,

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - Tavg < 350°F

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All safety injection pumps and centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated to be inoperable in accordance with the Surveillance Frequency Control Program while in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to ~~312°F,~~ MODE 5, or MODE 6 when the head is on the reactor vessel by either of the following methods: the POPS enable temperature specified in the PTLR,

a. By verifying that the motor circuit breakers have been removed from their electrical power supply circuits or,

b. By verifying that the pump is in a recirculation flow path and that two independent means of preventing RCS injection are utilized.

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

4. The number of tubes plugged during the inspection outage; and

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results.

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;

f. The results of any SG secondary side inspections.

Insert 2

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 1, 4, 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

Move to new page 6-24c

Attachment 2

Mark-up of Proposed Technical Specifications Bases Changes
(For Information Only)

Proposed Technical Specifications Bases Revised Pages

B3/4 4-1
B3/4 4-7
B 3/4 4-8
B 3/4 4-9
B 3/4 4-10
B 3/4 4-11
B 3/4 4-12
B3/4 4-13
B3/4 4-14
B3/4 4-15
B3/4 4-16
B3/4 4-17
B3/4 5-2

INSERT 1 (Bases Page 3/4 4-7)

In order to provide sufficient safety margins for protection against non-ductile failure, the pressure/temperature (P/T) limit curves for heatup and cooldown, inservice leak and hydrostatic testing, and criticality are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the 1/4T and 3/4T locations of the beltline and extended beltline materials for the applicability term stated in the PTLR. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

INSERT 2 (Bases Page 3/4 4-8)

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in UFSAR Section 5.2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. To ensure that the radiation embrittlement effects are accounted for in the calculation of the P/T limit curves, the most limiting RT_{NDT} includes the radiation induced shift, ΔRT_{NDT} corresponding to the end of the period for which curves are generated. This adjusted reference temperature (ART) can be predicted based upon the fluence and the copper and nickel content of the material in question. The ART is based upon the largest value of RT_{NDT} computed by the methodology presented in Regulatory Guide 1.99, Revision 2. The PTLR contains the results of the ART evaluations. The predicted neutron fluence, as a function of Effective Full Power Years (EFPY), has been calculated based on the guidance of Regulatory Guide 1.190. The PTLR contains the results of the fluence evaluations.

INSERT 3 (Bases Page 3/4 4-11)

The heatup and cooldown curves are composite curves, constructed based on a point-by-point comparison of the steady-state and finite rate data, as well as the reactor vessel and head flange, to identify the most restrictive curve. The use of the composite curve is necessary to set conservative heatup/cooldown limitations because at any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations have more restrictive limits, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve requirement is that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for inservice leak rate testing.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, meet the DNB design criteria during all normal operations and anticipated transients. In MODES 1 and 2 with less than all coolant loops in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal for removing decay heat; but, single failure considerations require all loops be in operation whenever the rod control system is energized and at least one loop be in operation when the rod control system is deenergized.

In MODE 4, a single reactor coolant loop or RHR loop provides sufficient heat removal for removing decay heat; but, single failure considerations require that at least 2 loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two RHR loops be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
- A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disabled

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that a single failure does not cause a loss of decay heat removal.

The operation of one Reactor Coolant Pump or one RHR Pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during Boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with Boron concentration reductions will, therefore, be within the capability of operator recognition and control.

the POPS enable temperature specified in the PTLR

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to ~~312°F~~ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer (thereby providing a volume into which the primary coolant can expand, or (2) by restricting the starting of Reactor Coolant Pumps to those times when secondary water temperature in each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rate (with the exception of the pressurizer) shall be limited in accordance with ~~Figures 3.4-2 and 3.4-3~~ those specified in the the PTLR for the service period specified ~~thereon~~ therein.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines ~~shown.~~ specified in the PTLR. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) ~~Figures 3.4-2 and 3.4-3 define limits to~~ The limits specified in the PTLR assure prevention of ~~nonductile~~ non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) ~~These limit lines~~ The limit lines specified in the PTLR shall be calculated periodically using methods provided ~~below.~~ in Technical Specification 6.9.1.11.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. ~~These properties are then evaluated in accordance with Appendix G of the 1996 Summer Addenda to Section XI of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.~~

~~Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.~~

Replace with Bases Insert 1

~~The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. An adjusted reference temperature, (ART), based upon the fluence and the copper and nickel content of the material in question, can be predicted.~~

~~The ART is based upon the largest value of RT_{NDT} computed by the methodology presented in Regulatory Guide 1.99, Revision 2. The ART for each material is given by the following expression:~~

$$\cancel{ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin}$$

~~Initial RT_{NDT} is the reference temperature for the unirradiated material. ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by the irradiation and is calculated as follows:~~

$$\cancel{\Delta RT_{NDT} = Chemistry\ Factor \times Fluence\ Factor}$$

~~The Chemistry Factor, CF (F), is a function of copper and nickel content. It is given in Table B3/4.4-2 for welds and in Table B3/4.4-3 for base metal (plates and forgings). Linear interpolation is permitted.~~

~~The predicted neutron fluence as a function of Effective Full Power Years (EFPY) has been calculated and is shown in Figure B3/4.4-1. The fluence factor can be calculated by using Figure B3/4.4-2. Also, the neutron fluence at any depth in the vessel wall is determined as follows:~~

$$\cancel{f = (f\ surface) \times (e^{-0.24X})}$$

~~where "f surface" is from Figure B3/4.4-1, and X (in inches) is the depth into the vessel wall.~~

~~Finally, the "Margin" is the quantity in °F that is to be added to obtain conservative, upper-bound values of adjusted reference temperature for the calculations required by Appendix G to 10 CFR 50.~~

$$\cancel{Margin = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}}$$

~~If a measured value of initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If generic value of initial RT_{NDT} is used, σ_I should be obtained from the same set of data. The standard deviations, for ΔRT_{NDT}, σ_Δ, are 28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT} surface.~~

~~The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY.~~

Replace with Bases Insert 2

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 ~~and these methods are discussed in detail in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999~~.

~~The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section XI as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature,~~ ~~RT_{NDT}~~~~, is used and this includes the radiation induced shift,~~ ~~ΔRT_{NDT}~~ ~~corresponding to the end of the period for which heatup and cooldown curves are generated.~~

~~The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor,~~ ~~K_I~~~~, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor,~~ ~~K_{IC}~~~~, for the metal temperature at that time.~~ ~~K_{IC}~~ ~~is obtained from the reference fracture toughness curve, defined in ASME Code Case N-640. The~~ ~~K_{IC}~~ ~~curve is given by the equation:~~

$$\cancel{K_{IC} = 33.2 + 20.734 \exp\,[0.02(T-RT_{NDT})]} \qquad \cancel{(1)}$$

~~where~~ ~~K_{IC}~~ ~~is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature~~ ~~RT_{NDT}~~~~. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:~~

$$\cancel{C\,K_{IM} + K_{IT} \leq K_{IC}} \qquad \cancel{(2)}$$

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~~where K_{IM} is the stress intensity factor caused by membrane (pressure) stress.~~

~~K_{IT} is the stress intensity factor caused by the thermal gradients.~~

~~K_{IC} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.~~

~~C = 2.0 for level A and B service limits, and~~

~~C = 1.5 for inservice hydrostatic and leak test operations.~~

~~At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and from these the allowable pressures are calculated.~~

~~COOLDOWN~~

~~For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.~~

~~The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IC} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IC} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.~~

~~The above procedures are needed because there is no direct control on temperature at the 1/4T location, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.~~

~~HEATUP~~

~~Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stress at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature. Therefore, the K_{IC} for the 1/4T crack during heatup is lower than the K_{IC} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IC}s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.~~

~~The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.~~

~~Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.~~

~~The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.~~

Replace with Bases Insert 3

~~Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Salem). Table B3/4.4-1 indicates that the limiting RT_{NDT} of 28°F occurs in the closure head flange of Salem Unit 2, and the minimum allowable temperature of this region is 148°F at pressures greater than 621 psig. These limits do not affect Figures 3.4-2 and 3.4-3.~~

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two POPSs valves or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to ~~312°F~~ the POPS enable temperature specified in the PTLR. Either POPS valve ~~has~~ adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of an Intermediate Head Safety Injection pump and its injection into a water solid RCS, or the start of a High Head Safety Injection pump in conjunction with a running Positive Displacement pump and its injection into a water solid RCS. The minimum electrical power sources required to assure POPS operability (based on POPS meeting the single failure criteria) consist of a normal (via offsite power) and an emergency (via batteries) power source for each train of POPS. Emergency diesel generators are not required for POPS to meet single failure criteria and therefore are not required for POPS OPERABILITY.

LCO 3.0.4.b is not applicable to an inoperable ~~LTOP system~~ POPS when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ~~LTOP system~~ POPS. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

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TABLE 3/4.4-1

SALEM UNIT 2 REACTOR VESSEL TOUGHNESS DATA

Component	Plate No. or Weld No.	Material Type	Cu (%)	Ni (%)	T (°F)	50 ft-lb 35 Mil Temp (°F)	RT (°F)	Average Upper Shelf Energy Normal to Principal Working Direction (ft-lb)	Principal Working Direction (ft-lb)
Closure Hd Dome	B4708	A533BCL1	0.11	0.70	-40	45*	-15*	82.5	127
Closure Hd Peel	B5007-3	A533BCL1	0.12	0.57	-20	15*	-20*	97*	149
Closure Hd Peel	B4707-1	A533BCL1	0.10	0.55	0	51*	0*	84*	129
Closure Hd Peel	B4707-3	A533BCL1	0.13	0.63	0	66*	6*	84*	129.5
Closure Hd Flng	B4702-1	A508CL2	-	0.68	28*	39*	28*	104*	160
Vessel Flange	B5001	A508CL2	-	0.70	12*	4*	12*	107*	164
Inlet Nozzle	B4703-1	A508CL2	-	0.69	60*	62*	60*	>72*	>111**
Inlet Nozzle	B4703-2	A508CL2	-	0.69	60*	25*	60*	>61*	>94**
Inlet Nozzle	B4703-3	A508CL2	-	0.68	60*	32*	60*	>71*	>109**
Inlet Nozzle	B4703-4	A508CL2	-	0.81	60*	40*	60*	80*	123.5
Outlet Nozzle	B4704-1	A508CL2	-	0.84	60*	8*	60*	82*	126
Outlet Nozzle	B4704-2	A508CL2	-	0.77	60*	20*	60*	75*	116
Outlet Nozzle	B4704-3	A508CL2	-	0.69	28*	8*	28*	82*	126
Outlet Nozzle	B4704-4	A508CL2	-	0.71	60*	40*	60*	77*	119
Upper Shell	B4711-1	A533BCL1	0.11	0.55	0*	50*	0*	87*	134
Upper Shell	B4711-2	A533BCL1	0.14	0.56	-10	60*	0*	79*	122
Upper Shell	B4711-3	A533BCL1	0.12	0.58	-10	88*	28*	69*	107
Inter. Shell	B4712-1	A533BCL1	0.13	0.56	0	<60	0	106	138
Inter. Shell	B4712-2	A533BCL1	0.12	0.62	-20	72	12	97	127.5
Inter. Shell	B4712-3	A533BCL1	0.11	0.57	-50	70	10	107	116
Lower Shell	B4713-1	A533BCL1	0.12	0.60	-10	68	8	98	127
Lower Shell	B4713-2	A533BCL1	0.12	0.57	-20	68	8	103	135.5
Lower Shell	B4713-3	A533BCL1	0.12	0.58	-10	70	10	121	135.5
Bottom Hd Peel	B4709-1	A533BCL1	0.12	0.60	-30	54*	-6*	90*	139
Bottom Hd Peel	B4709-2	A533BCL1	0.12	0.58	-20	42*	-18*	89*	137.5
Bottom Hd Peel	B4709-3	A533BCL1	0.11	0.56	-20	71*	11*	93*	143
Bottom Head	B4710	A533BCL1	0.12	0.60	-30	60*	0*	77*	118
Circum. Weld Bet Nozzle Shell & Int. Shell	8-442	-	0.28	0.74	-	-	-56***	-	-
Circum. Weld Bet Int. Shell & Lower Shell	9-442	-	0.197	0.060	-	-	-56***	99.7	-
Int. Shell Vertical Weld	2-442 [A,B,C]	-	0.219	0.735	-	-	-56***	96.2	-
Lower Shell Vertical Weld	3-442 [A,B,C]	-	0.213	0.867	-	-	-56***	114	-

* Estimated per NRC Standard Review Plan Section 5.8.2.

** 100% Shear not reached

*** Estimate per Pressurized Thermal Shock Rule, 10 CFR 50.61

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~~TABLE B 3/4.4.2~~

~~CHEMISTRY FACTOR FOR WELDS, °F~~

~~Copper, Wt-%~~	~~Nickel, Wt-%~~						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	[illegible]	[illegible]	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	[illegible]	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	[illegible]	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	[illegible]	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	173	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

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TABLE B 3/4.4-3

CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	[illegible]	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	[illegible]	317
0.40	175	189	207	231	257	288	320

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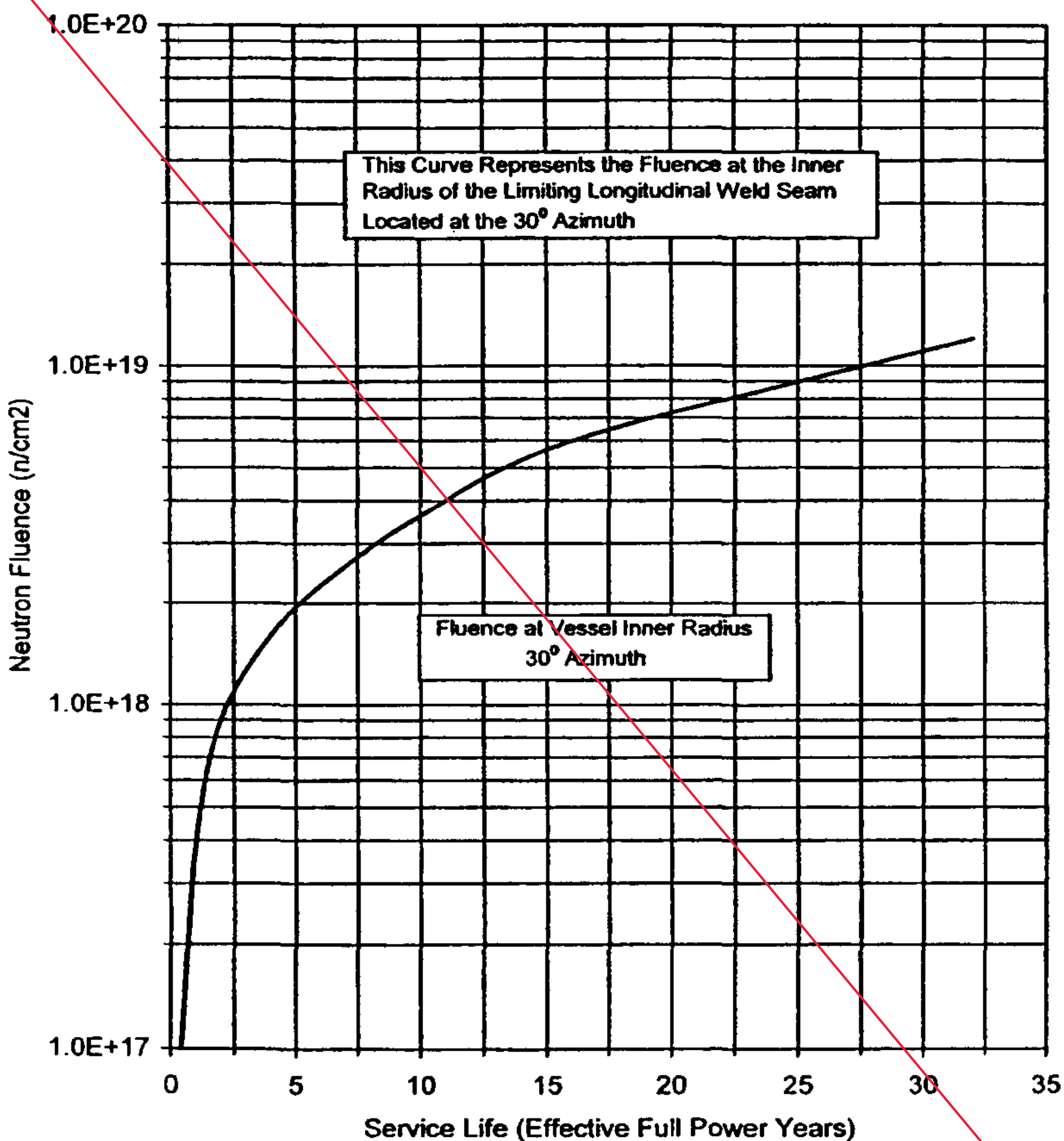

Figure B 3/4.4-1 Fast neutron fluence (E > 1 MeV) as a function of full power service life (EFPY)

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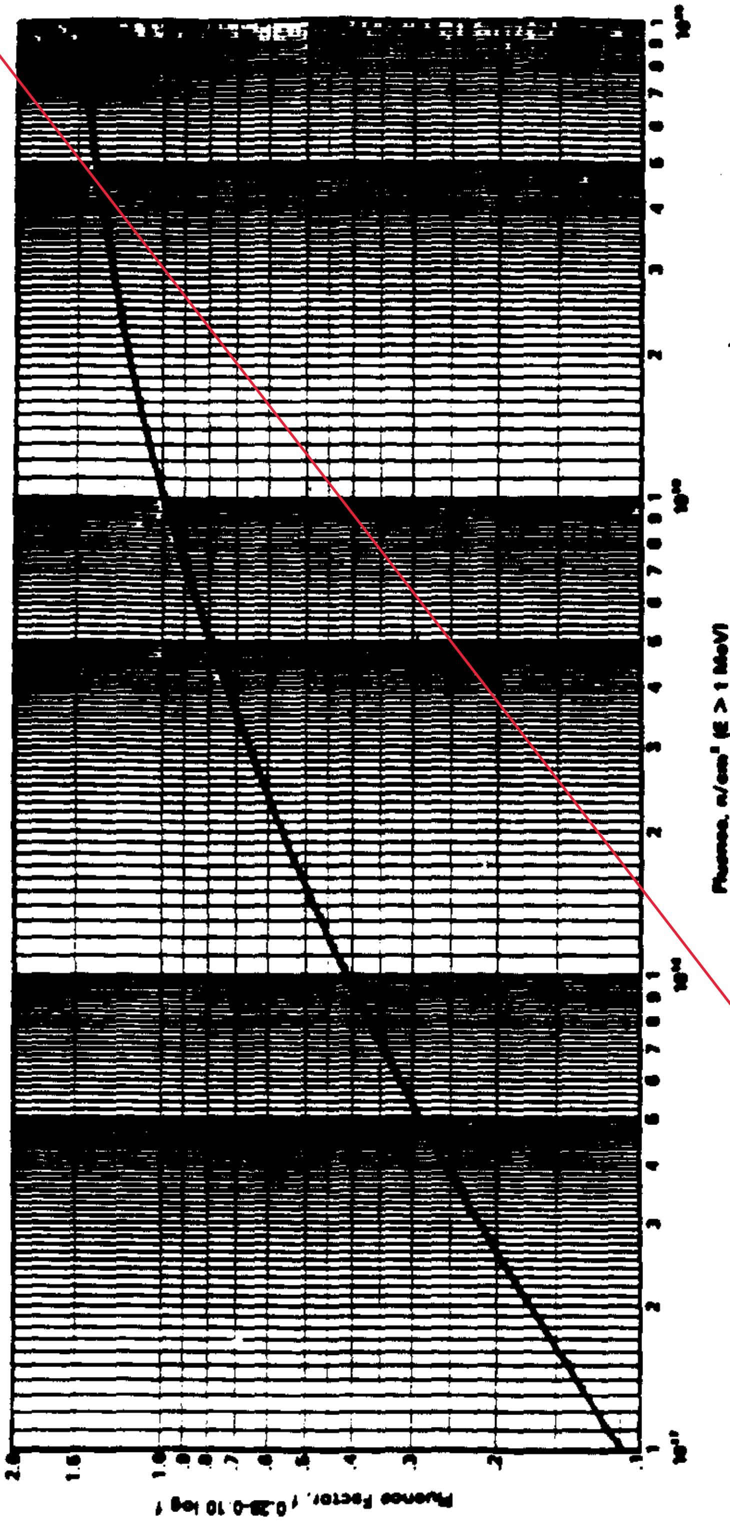

FIGURE B 3/4.4-2

Fluence Factor for use in the expression for ΔRT_{NDT}

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one safety injection pump or one centrifugal charging pump to be OPERABLE and the Surveillance requirement to verify all safety injection pumps except the allowed OPERABLE safety injection pump to be inoperable below ~~312°F~~ provides assurance that a mass addition pressure transient can be relieved by the operation of a single POPS relief valve.

the POPs enable temperature specified in the PTLR

When running a safety injection pump with the RCS temperature less than ~~312 °F~~ with the potential for injecting into the RCS and creating a mass addition pressure transient, two independent means of preventing reactor coolant system injection will be utilized. The two independent means can be satisfied by any of the following methods:
(1) A manual isolation valve locked in the closed position; or
(2) Two manual isolation valves closed; or
(3) One motor operated valve closed and its breaker de-energized and control circuit fuses removed; or
(4) One air operated valve closed and its air supply maintained in such a manner as to ensure that the valve will remain closed.

The surveillance requirements, which are provided to ensure the OPERABILITY of each component, ensure that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The safety analyses make the assumptions with respect to: 1) both the maximum and minimum total system resistance, and 2) both the maximum and minimum branch injection line resistance. These resistances, in conjunction with the ranges of potential pump performance, are used to calculate the maximum and minimum ECCS flow assumed in the safety analyses.

The maximum and minimum flow surveillance requirements in conjunction with the maximum and minimum pump performance curves ensures that the assumptions of total system resistance and the distribution of that system resistance among the various paths are met.

The maximum total pump flow surveillance requirements ensure the pump runout limits of 560 gpm for the centrifugal charging pumps and 675 gpm for the safety injection pumps are not exceeded. Due to the effect of pump suction boost alignment, the runout limits for the surveillance criteria are ≤ 554 gpm for C/SI pumps, ≤ 664 gpm for SI pumps in cold leg alignment and ≤ 654 gpm for SI pumps in hot leg alignment.

The surveillance requirement for the maximum difference between the maximum and minimum individual injection line flows ensure that the minimum individual injection line resistance assumed for the spilling line following a LOCA is met.

LCO 3.0.4.b is not applicable to an inoperable ECCS high head subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS high head subsystem. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

Attachment 3

Justification for Application of the WCAP-18124-NP-A Methodology for the Salem Unit 2 Fluence Determination

1.0 Introduction

RAPTOR-M3G has been used in the fast neutron (E > 1.0 MeV) fluence evaluation in support of updating reactor pressure vessel fluence for Salem Unit 2 to generate normal operation heatup and cooldown limit curves. Per the requirement of Regulatory Issue Summary (RIS) 2014-11 (Reference 1), the reactor pressure vessel (RPV) nozzles and any RPV materials that exceed a fast neutron (E > 1.0 MeV) fluence of 1E+17 n/cm^2 at the end-of-license extension (EOLE) must be evaluated with respect to fracture toughness. The RPV materials evaluated for Salem Unit 2 are listed in Table 1. Certain materials in this table were determined to have fluence values less than 1E+17 n/cm^2 and therefore did not require specific evaluation with respect to neutron embrittlement and fracture toughness.

The active core height extends from -182.88 cm to +182.88 cm in the RAPTOR-M3G model that represents the axial extension of the traditional beltline region of the reactor vessel which, by definition, is the region that directly surrounds the effective height of the active core (Reference 2). When compared to the axial elevations of the RPV materials evaluated in Table 1, the intermediate shell, intermediate shell longitudinal welds, intermediate shell to lower shell circumferential weld, lower shell, and lower shell longitudinal welds are subsumed under the traditional beltline region. Therefore, these materials have been approved by the United States Nuclear Regulatory Commission (USNRC) for generic application of RAPTOR-M3G for fast neutron (E > 1.0 MeV) fluence determination per WCAP-18124-NP-A (Reference 3).

The 50 effective full power years (EFPY) fast neutron (E > 1.0 MeV) fluence at the EOLE and locations for the RPV materials in the extended beltline region for Salem Unit 2 are listed in Table 2. The limitations and conditions for using RAPTOR-M3G for fast neutron (E > 1.0 MeV) fluence determination are stipulated in Section 4.0 of the safety evaluation letter captured in Reference 3. The limitations and conditions of Reference 3 are quoted below:

1. *Applicability of WCAP-18124-NP, Revision 0, is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to response parameters of interest (e.g., pressure- temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and reactor coolant system inlet and outlet nozzles and reactor vessel internal components.*

2. *Least squares adjustment is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the inconsistency should be disqualified.*

The second limitation and condition listed above does not apply as the least-squares procedures were not used to adjust the calculated fast neutron (E > 1.0 MeV) fluence values for RPV materials evaluated in the reactor vessel integrity analysis. The least-squares results were only used to compare the calculations and measurements from the evaluated dosimetry and validate the neutron transport models, and those comparisons showed satisfactory results.

The Reference 3 neutron fluence methodology, however, has been used to determine the fast neutron (E > 1.0 MeV) fluence values for RPV materials in the extended beltline region. Therefore, additional justification should be provided for use of the methodology in this region. The following information provides the additional justification of using RAPTOR-M3G for fast neutron (E > 1.0 MeV) fluence determination for the RPV extended beltline regions at Salem Unit 2 by summarizing the additional benchmark data from a Westinghouse 4-loop pressurized water reactor (PWR). In addition, WCAP-18124-NP-A, Rev. 0, Supplement 1-NP-A Rev. 0 (ML22153A139) has been approved, which generically approves WCAP-18124-NP-A methodology for use in the extended beltline.

Table 1: Locations for Reactor Vessel Materials

Material	Azimuthal Location[1] [°] Min	Azimuthal Location[1] [°] Max	Axial Location[2] [cm] Min	Axial Location[2] [cm] Max
Outlet Nozzle to Nozzle Shell Welds[3]				
- Nozzle 1	22.0		270.12	
- Nozzle 2	158.0		270.12	
- Nozzle 3	202.0		270.12	
- Nozzle 4	338.0		270.12	
Inlet Nozzle to Nozzle Shell Welds[3]				
- Nozzle 1	67.0		265.04	
- Nozzle 2	113.0		265.04	
- Nozzle 3	247.0		265.04	
- Nozzle 4	293.0		265.04	
Nozzle Shell				
- Plate 1	300.0	60.0	233.53	485.07
- Plate 2	60.0	180.0	233.53	485.07
- Plate 3	180.0	300.0	233.53	485.07
Nozzle Shell Longitudinal Welds[3]				
- Weld 1	60.0		233.53	485.07
- Weld 2	180.0		233.53	485.07
- Weld 3	360.0		233.53	485.07
Nozzle Shell to Intermediate Shell Circumferential Weld – Centerline[3]	0.0	360.0	233.53	
Intermediate Shell				
- Plate 1	0.0	120.0	-42.93	233.53
- Plate 2	120.0	240.0	-42.93	233.53
- Plate 3	240.0	360.0	-42.93	233.53
Intermediate Shell Longitudinal Welds[3]				
- Weld 1	0.0		-42.93	233.53
- Weld 2	120.0		-42.93	233.53
- Weld 3	240.0		-42.93	233.53
Intermediate Shell to Lower Shell Circumferential Weld – Centerline[3]	0.0	360.0	-42.93	
Lower Shell				
- Plate 1	300.0	60.0	-313.52	-42.93
- Plate 2	60.0	180.0	-313.52	-42.93
- Plate 3	180.0	300.0	-313.52	-42.93
Lower Shell Longitudinal Welds[3]				
- Weld 1	60.0		-313.52	-42.93
- Weld 2	180.0		-313.52	-42.93
- Weld 3	360.0		-313.52	-42.93
Lower Shell to Lower Vessel Head Circumferential Weld – Centerline[3]	0.0	360.0	-313.52	

Notes:
1. Azimuthal locations are indexed to θ = 0.0 as shown on reactor pressure vessel general assembly drawing.
2. Axial elevations are indexed to Z = 0.0 at the midplane of the active fuel stack.
3. Longitudinal welds are modelled to 3.49 cm width. Circumferential welds are modelled using a 3.18 cm axial width.

Table 2: Fast Neutron (E > 1.0 MeV) Fluence at 50 EFPY and Locations for RPV Extended Beltline Region Materials

Material	Axial Location[1] [cm]	Unit 2 [n/cm^2]
Outlet Nozzle to Nozzle Shell Welds - Lowest Extent	270.12	1.78E+16
Inlet Nozzle to Nozzle Shell Welds - Lowest Extent	265.04	2.26E+16
Nozzle Shell	233.53	2.44E+17
Nozzle Shell Longitudinal Welds	233.53	1.83E+17
Nozzle Shell to Intermediate Shell Circumferential Weld[2]	233.53	2.80E+17
Lower Shell to Lower Vessel Head Circumferential Weld[2]	-313.52	3.24E+15

Notes:

1. Values listed are indexed to Z = 0.0 at the midplane of the active fuel stack and only the closest distance to the core midplane is listed.
2. The fluence values for these welds are reported at the maximum location within the width.

2.0 Additional Benchmarking Measurements

In order to collect measurement benchmark data for the extended beltline region, three sets of ex-vessel neutron dosimetry (EVND) have been installed at the elevation of the reactor vessel support for a Westinghouse 4-loop plant. The elevation of the reactor vessel support is approximately 8.5 feet above the core midplane. The specific axial locations of the EVND capsules to the core midplane (Z = 0.0 cm) and times of irradiation are listed in Table 3. The dosimeter foils included in these EVND capsules are listed in Table 4. The measured dosimetry reactions for those foils are listed in Table 5.

Table 3: Location and Time of Irradiation for Sensor Sets Analyzed at RPV Supports

Capsule ID	Sensor Location	Azimuthal Location	Axial Elevation [cm]	Cycle(s) of Irradiation
E	Ex-Vessel	180°	257.99	11
A	Ex-Vessel	225°	255.75	11
K	Ex-Vessel	180°	257.99	12 – 19
M	Ex-Vessel	180°	257.99	20
R	Ex-Vessel	135°	261.65	20

Table 4: Foil Sensor Set Contents in EVND at RPV Supports

Capsule ID	Radiometric Monitor Foils							
	Fe	Ni	Cu	Ti	Co	Nb	U-238	Np-237
E	x	x	X	x	x		x	x
A	x	x	X	x	x		x	x
K	x	x	X	x	x	x		
M	x	x	X	x	x	x		
R	x	x	X	x	x	x		

Table 5: Measured Dosimetry Reactions in EVND at RPV Supports

Material	Reaction of Interest	Neutron Energy Response(1)	Product Half-Life(2)
Copper	^{63}Cu (n,α) ^{60}Co	4.53-11.0 MeV	5.271 y
Titanium	^{46}Ti (n,p) ^{46}Sc	3.70-9.43 MeV	83.788 d
Iron	^{54}Fe (n,p) ^{54}Mn	2.27-7.54 MeV	312.13 d
Nickel	^{58}Ni (n,p) ^{58}Co	1.98-7.51 MeV	70.86 d
238U	^{238}U (n,f) ^{137}Cs	1.44-6.69 MeV	30.05 y
Niobium	^{93}Nb (n,n′) ^{93m}Nb	0.95-5.79 MeV	16.13 y
237Np	^{237}Np (n,f) ^{137}Cs	0.68-5.61 MeV	30.05 y
Cobalt - Al	^{59}Co (n,g) ^{60}Co	Thermal	5.271 y

Note(s):

(1) Energies between which 90% of activity is produced (^{235}U fission spectrum). Ref. ASTM E844-18.
(2) Half-life data is from ASTM E1005-16.

2.1 <u>Additional Benchmarking Neutron Transport Calculations</u>

The neutron transport calculations for the additional benchmarking at the 4-loop Westinghouse plant extended beltline region followed the Westinghouse fluence methodology described in Reference 3, which is the same methodology used for the neutron transport calculations performed in support of Salem Unit 2 reactor pressure vessel fluence evaluation work. In the application of this methodology to the fast neutron exposure evaluations for the 4-loop Westinghouse plant EVND dosimetry sets at the RPV supports, forward transport calculations were carried out to directly solve for the space- and energy-dependent scalar flux, $\varphi(r,\theta,z,E)$.

For the additional benchmark analysis, all of the transport calculations were carried out using the RAPTOR-M3G three-dimensional discrete ordinates code and the BUGLE-96 (Reference 4) cross-section library. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray group cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modelled with an S_{16} order of angular quadrature.

A plan view of the reactor model is shown in Figure 1. In addition to the core, reactor internals, RPV, and concrete bioshield, the model also included explicit representations of the surveillance capsules, RPV clad, and RPV nozzles and supports. Section views of the reactor model are shown in Figure 2 and Figure 3.

In developing the model of the reactor geometry, nominal design dimensions were used for the various structural components. Water temperatures (and densities) in the core, bypass, and downcomer regions of the reactor were taken to be representative of full-power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The r,θ,z geometric mesh description of the reactor model consisted of 233 radial by 186 azimuthal by 469 axial mesh intervals. Mesh sizes were chosen to ensure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion used in the calculations was 0.001.

The core power distributions used in the additional benchmarking transport analysis included fuel-assembly-specific initial enrichments, burnups, and axial power distributions. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of the fuel cycle-averaged neutron fluence rate, which, when multiplied by the appropriate fuel cycle length, provide the incremental fast neutron ($E > 1.0$ MeV) fluence exposure for each fuel cycle. The energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial ^{235}U enrichment and burnup history of the individual fuel assemblies. From the assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined. These fuel-assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the r,θ,z spatial mesh arrays used in the RAPTOR-M3G discrete ordinates calculations.

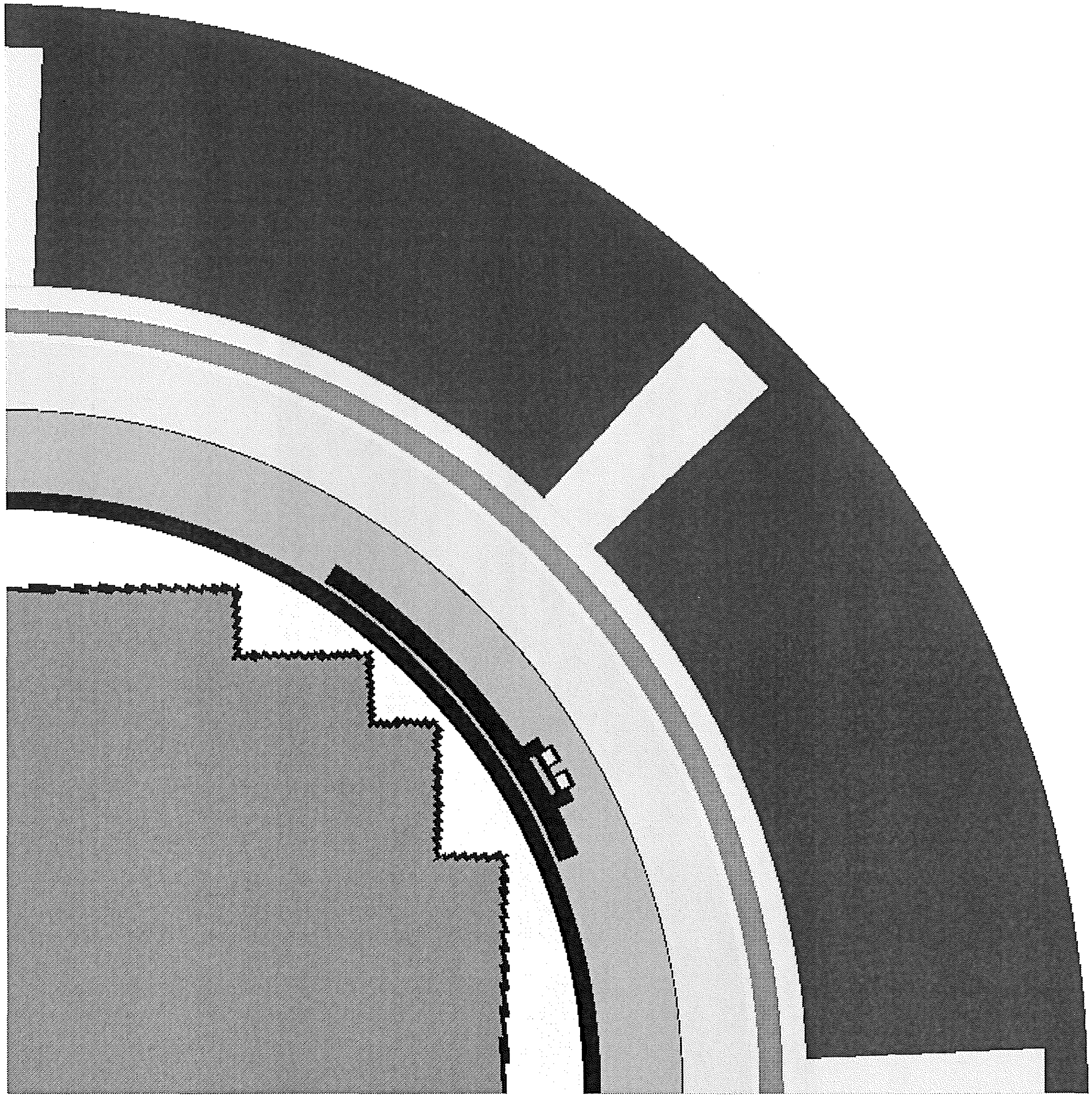

Figure 1: Reactor Geometry – Plan View at Core Midplane

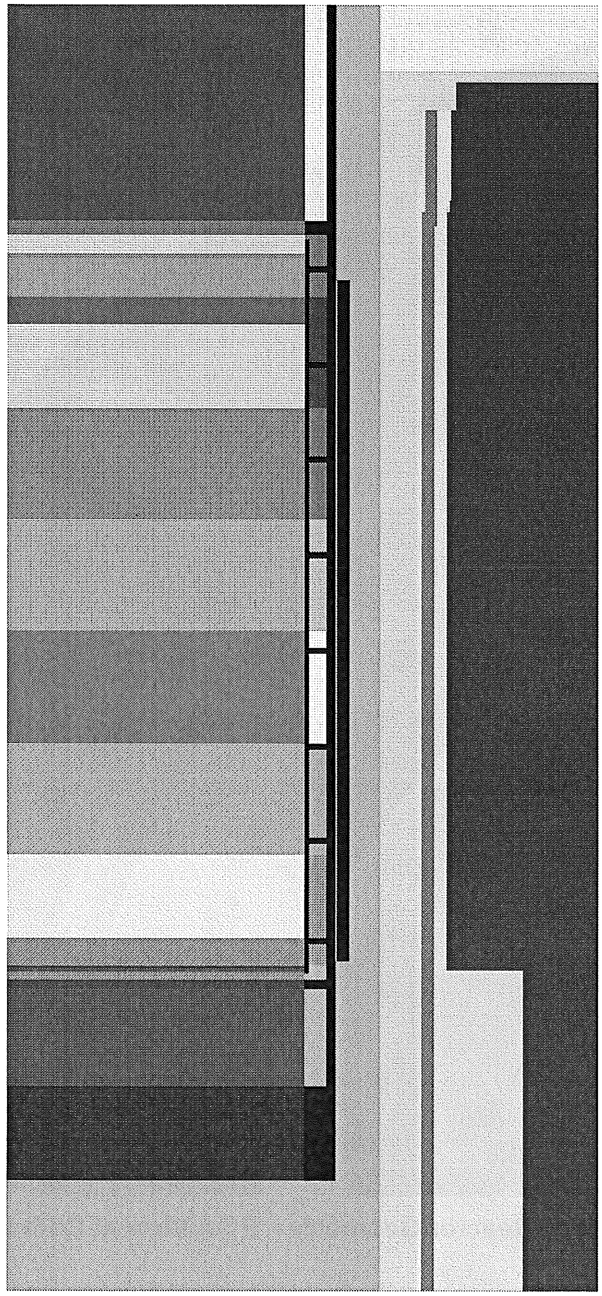

Figure 2: Reactor Geometry – Section View at Outlet Nozzle Centerline

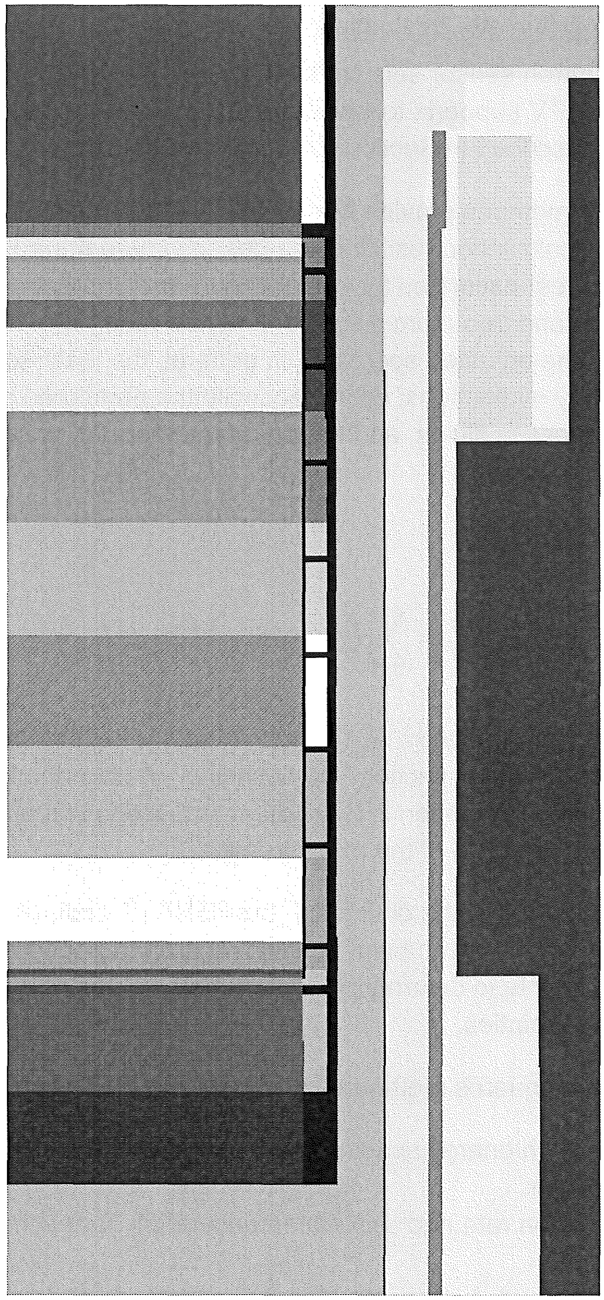

Figure 3: Reactor Geometry – Section View at Inlet Nozzle Centerline

2.2 Additional Benchmarking Dosimetry Evaluations

The evaluations of the neutron sensor sets contained in the EVND dosimetry capsules at the 4-loop Westinghouse plant RPV supports followed the state-of-the-art least-squares dosimetry evaluation methodology described in Section 3.0 of Reference 3.

Least-squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best-estimates for key exposure parameters such as fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to reactor dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties.

For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})\ (\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, φ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the dosimetry, the FERRET code (Reference 5) was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters along with associated uncertainties.

The application of the least-squares methodology requires the following input.

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.
3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the current application, the calculated neutron spectrum at each measurement location was obtained from the results of the previously described additional benchmarking neutron transport calculations. The spectrum at each sensor set location was input in an absolute sense (rather than simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements. The sensor reaction rates were derived from the measured specific activities of each sensor set and the operating history of the respective fuel cycles. The dosimetry reaction cross sections were obtained from the SNLRML dosimetry cross section library (Reference 6).

In addition to the magnitude of the calculated neutron spectra, the measured sensor set reaction rates, and the dosimeter set reaction cross sections, the least-squares procedure requires uncertainty estimates for each of these input parameters. The following provides a summary of the uncertainties associated with the least-squares evaluation of the dosimetry.

2.3 Additional Benchmarking Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM International consensus standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input into the least-squares evaluation:

Reaction	Uncertainty
^{63}Cu (n,α) ^{60}Co	5%
^{46}Ti (n,p) ^{46}Sc	5%
^{54}Fe (n,p) ^{54}Mn	5%
^{58}Ni (n,p) ^{58}Co	5%
^{238}U (n,f) ^{137}Cs	10%
^{93}Nb (n,n′)^{93m}Nb	10%
^{237}Np (n,f) ^{137}Cs	10%
^{59}Co (n,γ) ^{60}Co	35%[1]

These uncertainties are given at the 1σ level.

[1] The cobalt content of older Co-Al foils used in EVND is not known for certain, but is believed to be between 0.438% and 0.562%. To account for this unknown, the uncertainty assigned in the least-squares evaluations (typically 5%) was increased by roughly (0.562 / 0.438 = 1.28) to 5% + 28% = 33%. Rounded to the nearest five, an uncertainty of 35% was input.

2.4 Additional Benchmarking Dosimetry Cross Section Uncertainties

As previously noted, the reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multi-group structure for use in least-squares adjustment applications. These cross sections were compiled from the ENDF/B-VI cross section evaluations and have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors are provided in Reference 6.

For sensors included in the dosimetry sets, the following uncertainties in the fission spectrum-averaged cross sections are provided in the SNLRML documentation package:

Reaction	Uncertainty
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	4.08-4.16%
$^{46}Ti\ (n,p)\ ^{46}Sc$	4.51-4.87%
$^{54}Fe\ (n,p)\ ^{54}Mn$	3.05-3.11%
$^{58}Ni\ (n,p)\ ^{58}Co$	4.49-4.56%
$^{238}U\ (n,f)\ ^{137}Cs$	0.54-0.64%
$^{93}Nb\ (n,n')^{93m}Nb$	6.96-7.23%
$^{237}Np\ (n,f)\ ^{137}Cs$	10.32-10.97%
$^{59}Co\ (n,\gamma)\ ^{60}Co$	0.76-3.59%

These tabulated ranges provide an indication of the dosimetry cross section uncertainties associated with the sensor sets used in LWR irradiations.

2.5 Additional Benchmarking Calculated Neutron Spectrum Uncertainties

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks, and the dosimetry cross section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

Where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties $R_{g'}$ and R_g specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = (1-\theta)\delta_{gg'} + \theta e^{-H}$$

Where:

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g' and 0.0 otherwise.

The set of parameters defining the input covariance matrix for calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short-Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

These uncertainty assignments are consistent with an industry consensus uncertainty of 15 – 20% (1σ) for the fast neutron portion of the spectrum and provide for a reasonable increase in the uncertainty for neutrons in the intermediate and thermal energy ranges.

2.6 <u>Additional Benchmarking Measurement-to-Calculation Comparison</u>

The comparison of the measurement results from each of the sensor set irradiations at RPV supports with corresponding analytical predictions at the measurement locations are presented in Table 6 and Table 7. These comparisons are provided on two levels. On the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. On the second level, calculated values of neutron exposure rates in terms of fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table 6, comparisons of measurement-to-calculation (M/C) ratios are listed for the threshold sensors contained in the EVND dosimetry capsules irradiated at RPV supports that are approximately 8.5 feet above the core midplane. For the individual threshold foils, the average M/C ratio ranges from 0.49 to 1.33, with an overall average of 0.75 and an associated standard deviation of 28%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table 7, best-estimate-to-calculation (BE/C) ratios for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate resulting from the least-squares evaluation of the dosimetry sets is provided for the EVND capsules irradiated at the RPV supports, which are approximately 8.5 feet above the core midplane. The BE/C ratio for the fast neutron (E > 1.0 MeV) fluence rate is 0.79 with an associated standard deviation of 19% and 0.88 with an associated standard deviation of 20% for the iron atom displacement rate (dpa/s). These BE/C ratios are within the ± 20% uncertainty at 1-σ level required by Regulatory Guide (RG) 1.190 (Reference 7).

In summary, for the extended beltline region, the M/C data provided in Table 6 as well as the BE/C data provided in Table 7 suggest that the calculations are over predicting the neutron exposure, particularly at the high end of the energy spectrum. For instance, the bottom of the 90% neutron response for the copper, titanium, iron, and nickel dosimeters is 4.53 MeV, 3.70 MeV, 2.27 MeV, and 1.98 MeV, respectively. Neutrons with energies greater than these constitute a small fraction of the neutron (E > 1.0 MeV) fluence rate in the extended beltline region. The BE/C values in Table 7 account for the spectral coverage of the different sensors, and provide an estimate of the key damage parameters, fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, that result from an uncertainty-weighted reconciliation of all of the measurements and calculations. The BE/C values in Table 7 suggest that the calculated damage parameters are moderately conservative relative to the best-estimate values.

Table 6: Measured-to-Calculated (M/C) Reaction Rates – Ex-Vessel Capsule Located in the Vicinity of the RPV Supports

Reaction	Capsule E	Capsule A	Capsule K	Capsule M	Capsule R	Average	% Std. Dev.
^{63}Cu (n,α) ^{60}Co	0.68	-	0.60	0.60	0.53	0.60	10%
^{46}Ti (n,p) ^{46}Sc	0.76	0.65	0.69	0.73	0.59	0.68	10%
^{54}Fe (n,p) ^{54}Mn	0.76	0.67	0.66	0.63	0.49	0.64	15%
^{58}Ni (n,p) ^{58}Co	0.78	0.68	0.68	0.74	0.54	0.68	13%
^{238}U(Cd) (n,f) ^{137}Cs	1.09	0.92	-	-	-	1.01	12%
^{93}Nb (n,n′) ^{93m}Nb	-	-	1.33	1.02	0.58	0.98	39%
^{237}Np(Cd) (n,f) ^{137}Cs	1.18	0.82	-	-	-	1.00	25%
Average of M/C Results						0.75	28%

Table 7: Best-Estimate-to-Calculated (BE/C) Exposure Rates – Ex-Vessel Capsule Located in the Vicinity of the RPV Supports

Capsule	Neutron (E > 1.0 MeV) Fluence Rate BE/C	Iron Atom Displacement Rate BE/C
E	0.95	1.03
A	0.78	0.84
K	0.87	1.03
M	0.82	0.91
R	0.55	0.61
Average	0.79	0.88
% Std. Dev.	19%	20%

3.0 <u>Justification Conclusions</u>

The uncertainty associated with a fluence determination methodology is comprised of two major components: the results of an analytic uncertainty analysis and the results of benchmarking comparisons. An analytic uncertainty analysis assesses the level of confidence in key input parameters to a fluence calculation and quantifies the impact that plausible input parameter variations have on calculated fluence results. Benchmarking comparisons refer to comparisons of fluence calculations performed with a candidate methodology to alternate calculations or to measurements from a representative environment.

A comprehensive analytical uncertainty analysis for the Salem Unit 2 RPV extended beltline region demonstrates that the RAPTOR-M3G based calculations have a maximum of 63% uncertainty for the reported fast neutron fluence in the RPV extended beltline region of Salem Unit 2, which is associated with the lower shell to lower vessel head circumferential weld. Note that the uncertainty is increasing as the axial elevation of the RPV materials moving away from the active core. For elevations that are not as far from the top and bottom of the active core, the uncertainty of the fast neutron (E > 1.0 MeV) fluence determined using RAPTOR-M3G is much less than the maximum uncertainty stated above. Also, the uncertainty is much larger for the fast neutron fluence values reported at the outer radius of the RPV wall than those reported at the inner surface of the RPV wall. For example, the estimated uncertainty for the nozzle shell and nozzle shell to intermediate shell circumferential weld is approximately 30%. The Analytical Uncertainty for the Salem Unit 2 extended beltline materials are summarized in Table 8. This completes an important part of qualifying RAPTOR-M3G as the fluence determination methodology for RPV extended beltline region per RG 1.190 (Reference 7) regulatory position 1.4.1.

Table 8: Fast Neutron (E > 1.0 MeV) Fluence at 50 EFPY and Associated Analysis Uncertainty for the Extended Beltline Region Materials

Material	Axial Location to Core Midplane [cm]	Unit 2 [n/cm^2]	Analytical Uncertainty
Outlet Nozzle to Nozzle Shell Welds - Lowest Extent	270.12	1.78E+16	35%
Inlet Nozzle to Nozzle Shell Welds - Lowest Extent	265.04	2.26E+16	35%
Nozzle Shell	233.53	2.44E+17	30%
Nozzle Shell Longitudinal Welds	233.53	1.83E+17	30%
Nozzle Shell to Intermediate Shell Circumferential Weld	233.53	2.80E+17	30%
Lower Shell to Lower Vessel Head Circumferential Weld	-313.52	3.24E+15	63%

Additional benchmarking work described in Section 2.0 was performed at a 4-loop Westinghouse plant near the RPV supports that are approximately 8.5 feet above the core midplane. This work concluded that the RAPTOR-M3G fluence determination methodology has about a 30% uncertainty in the fast neutron (E > 1.0 MeV) determination and the calculations typically overestimate the fast neutron (E > 1.0 MeV) fluence and iron atom displacement (dpa) at the extended beltline region. This completes the second important part of qualifying RAPTOR-M3G as the fluence determination methodology for RPV extended beltline region per RG 1.190 (Reference 7) regulatory position 1.4.2.

The RPV extended beltline materials evaluated for Salem Unit 2 in Table 2 are all located within an axial distance of 8.86 feet above or below the core midplane. However, the lower shell to lower vessel head circumferential weld is approximately 1.42 feet (43.47 cm) further away. It is important to note that this material is not classified as an extended beltline material since the projected fluence values for Salem Unit 2 are well below 1E+17 n/cm^2 and therefore not evaluated with respect to fracture toughness. This circumferential weld has a calculated fast neutron (E > 1.0 MeV) fluence of 3.24E+15 n/cm^2 for Salem Unit 2 at 50 EFPY using RAPTOR-M3G, which is more than a factor of 30 lower than the prescribed threshold of 1E+17 n/cm^2 for the definition of extended beltline region. Because the RAPTOR-M3G fluence determination methodology has a maximum of 63% uncertainty for the RPV extended beltline region this circumferential weld does not need to be included as extended beltline material that has to be evaluated for fracture toughness embrittlement effect.

From the discussion above, the methodology uncertainty for fast neutron (E > 1.0 MeV) fluence determination for these RPV extended beltline materials is also less than or equal to 63% for the fast neutron fluence values reported for Salem Unit 2. In review of downstream reactor vessel integrity analysis, significant margins in the fast neutron fluence on the extended beltline materials have been identified. For example, a review of RT_{PTS} of the extended beltline materials determined the fluence would need to increase by 40 times in the Nozzle Shell (i.e., Upper Shell B4711-3) for it to become limiting. A review of 3/4T ART of the extended beltline materials determined the fluence would need to increase by 6 times in the Nozzle Shell (i.e., Upper Shell B4711-3) for it to becoming limiting. The estimated uncertainty using RAPTOR-M3G for the Nozzle Shell elevation is less than 30%. Therefore, it is not credible that any extended beltline region materials would ever become limiting for Salem Unit 2 at 50 EFPY. Finally, Section 3.4 of Reference 8 states that unless the fast neutron (E > 1.0 MeV) fluence for the nozzle material is greater than 4.28E+17 n/cm^2, embrittlement need not be considered for nozzle forging evaluation and the nozzles will be non-limiting compared to the beltline with respect to the pressure-temperature limit curves. Embrittlement was conservatively considered in the reactor vessel integrity analyses for nozzle materials, even if the fluence was below this threshold. The fast neutron (E > 1.0 MeV) fluence values reported for both the inlet and outlet nozzles in Table 2 are more than a factor of 18 lower than this fast neutron (E > 1.0 MeV) fluence threshold. As the evaluated net RAPTOR-M3G methodology uncertainty is approximately 35% or less for this elevation based on additional benchmarking and analytical uncertainty analysis, it is also not credible that the inlet and outlet nozzle fast neutron (E > 1.0 MeV) fluence at the EOLE will exceed 4.28E+17 n/cm^2.

4.0 Application of WCAP-18124-NP-A, Revision 0 to the Extended Beltline Region – Conclusion

Limitation and Condition #1 has been addressed in that the additional benchmarking at the RPV extended beltline region summarized herein, margin assessment documented in Reference 8, and the fluence analysis have provided additional justification supporting the use of the Reference 3 methodology for the extended beltline regions of the Salem Unit 2 RPV.

References:

1. Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” October 14, 2014. (Agencywide Documents Access and Management System Accession Number ML14149A165)

2. 10 CFR Appendix G to Part 50, “Fracture Toughness Requirements.”
3. Westinghouse Report, WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018.
4. RSICC Data Library Collection DLC-185, “BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” March 1996. (Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.)
5. RSICC Computer Code Collection PSR-145 “FERRET: Least-Squares Solution to Nuclear Data and Reactor Physics Problems,” January 1980. (Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.)
6. RSICC Data Library Collection DLC-178, “SNLRML Recommended Dosimetry Cross Section Compendium,” July 1994. (Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.)
7. U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.
8. Pressurized Water Reactor Owners Group (PWROG) Report, PWROG-15109-NP-A, Revision 0, “PWR Pressure Vessel Nozzle Appendix G Evaluation,” January 2020.

Enclosure 2

Salem Unit 2 Pressure and Temperature Limits Report (PTLR)

PSEG Nuclear LLC

Salem Generating Station Unit 2

Pressure and Temperature Limits Report (PTLR) for 50 Effective Full-Power Years (EFPY)

Revision 0

Prepared by: ______________________ Date: 7-12-22
Samuel E. Speer
[Corporate Engineering Programs]

Reviewed by: A.A. FAKHAR - PER ATTACHED EMAIL Date: 7-21-22
Ali A. Fakhar
[Manager, Corporate Engineering Programs]

Approved by: ______________________ Date: 7-21-2022
William J. Kopchick
[Sr. Director, Engineering Services]

Speer, Samuel E.

From: Fakhar, Ali A.
Sent: Thursday, July 21, 2022 7:29 AM
To: Speer, Samuel E.
Subject: Salem Unit 2 Pressure Temperature Limit Report (PTLR)

Sam,

I have reviewed and approved the Salem Unit 2 Pressure-Temperature Limits Report.

Thanks,
Ali

Table of Contents

1.0 Purpose

The purpose of the Salem Generating Station Unit 2 (Salem Unit 2) Pressure and Temperature (P-T) Limits Report (PTLR) is to present operating limits relating to:

1. Reactor Coolant System (RCS) P-T limits during Heat-up and Cooldown, inservice hydrostatic testing, and criticality;
2. RCS Heatup and Cooldown rates;
3. Reactor Pressure Vessel (RPV) head flange boltup temperature limits, and
4. Pressurizer Overpressure Protection System (POPS) setpoints and enable temperature.

This PTLR summarizes the various technical methodologies, equations, calculations, etc. that were necessary to produce the P-T Curves and POPS setpoints and enable temperature. Throughout this PTLR, references will be made to the Westinghouse P-T limits analyses [1, 2] that consist of the detailed explanations, figures, and tables.

This PTLR has been prepared in accordance with the requirements of Technical Specification 6.9.1.11.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59 [10]. The revised PTLR shall be submitted to the NRC upon issuance.

2.0 Applicability

This report is applicable to the Salem Unit 2 RPV for up to 50 Effective Full-Power Years (EFPY).

The following Salem Unit 2 Technical Specifications (TS) are affected by the information contained in this report:

TS 3.1.2.3	Charging Pump – Shutdown
TS 3.4.1.3	Reactor Coolant Loops – Hot Shutdown
TS 3.4.1.4	Reactor Coolant Loops – Cold Shutdown
TS 3.4.10.1	Reactor Coolant System Pressure/Temperature Limits
TS 3.4.10.3	Reactor Coolant System Overpressure Protection Systems
TS 3.5.3	ECCS Subsystems – Tavg < 350°F

3.0 Operating Limits

All limits are valid until 50 EFPY, which is projected to be beyond the expiration of the operating license for Salem Unit 2.

Pressurizer Overpressurization Overpressure Protection System (POPS) Enable Temperature

312°F*

*The analyzed setpoint is 292°F, including instrument uncertainty, however, any changes to the setpoint of 312°F need to be evaluated under the Design Change Control Process.

Referenced in: TS 3.1.2.3, TS 3.4.10.1, TS 3.4.1.3, TS 3.4.1.4, TS 3.4.10.3, TS 3.5.3, and SR 4.5.3.2

RCS Pressure/Temperature (P/T) Limits

Figure 3-1* RCS P-T Limits for Heatup

Figure 3-2* RCS P-T Limits for Cooldown

Referenced in: 3.4.10.1, SR 4.4.10.1.1, and SR 4.4.10.1.2

*Figures include instrumentation uncertainty

(18°F temperature margin and 61 psi pressure margin)

NOTE: Tables B-1, B-2, and B-3 of Appendix B contain a tabulated version of the curves.

RCS Heatup/Cooldown Rate Limits

Maximum heatup of 100°F in any one hour period.

Maximum cooldown of 100°F in any one hour period, and

Maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic testing operations above system design pressure.

Referenced in: TS 3.4.10.1, SR 4.4.10.1.1, and SR 4.4.10.1.2

Pressurizer Overpressurization Overpressure Protection System (POPS) Setpoint

375 psig*

Referenced in: TS 3.4.10.3

*The analyzed setpoint is 434 psig, including instrument uncertainty, however, any changes to the setpoint of 375 psig need to be evaluated under the Design Change Control Process.

Minimum Boltup Temperature

62 °F*

Referenced in: PTLR Figures 3-1 and 3-2.

*This temperature value accounts for 2°F temperature margin, Section 6.2 of [1].

LIMITING MATERIAL AND ART VALUES AT 50 EFPY:
1/4T ART: 209°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)
3/4T ART: 150°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)

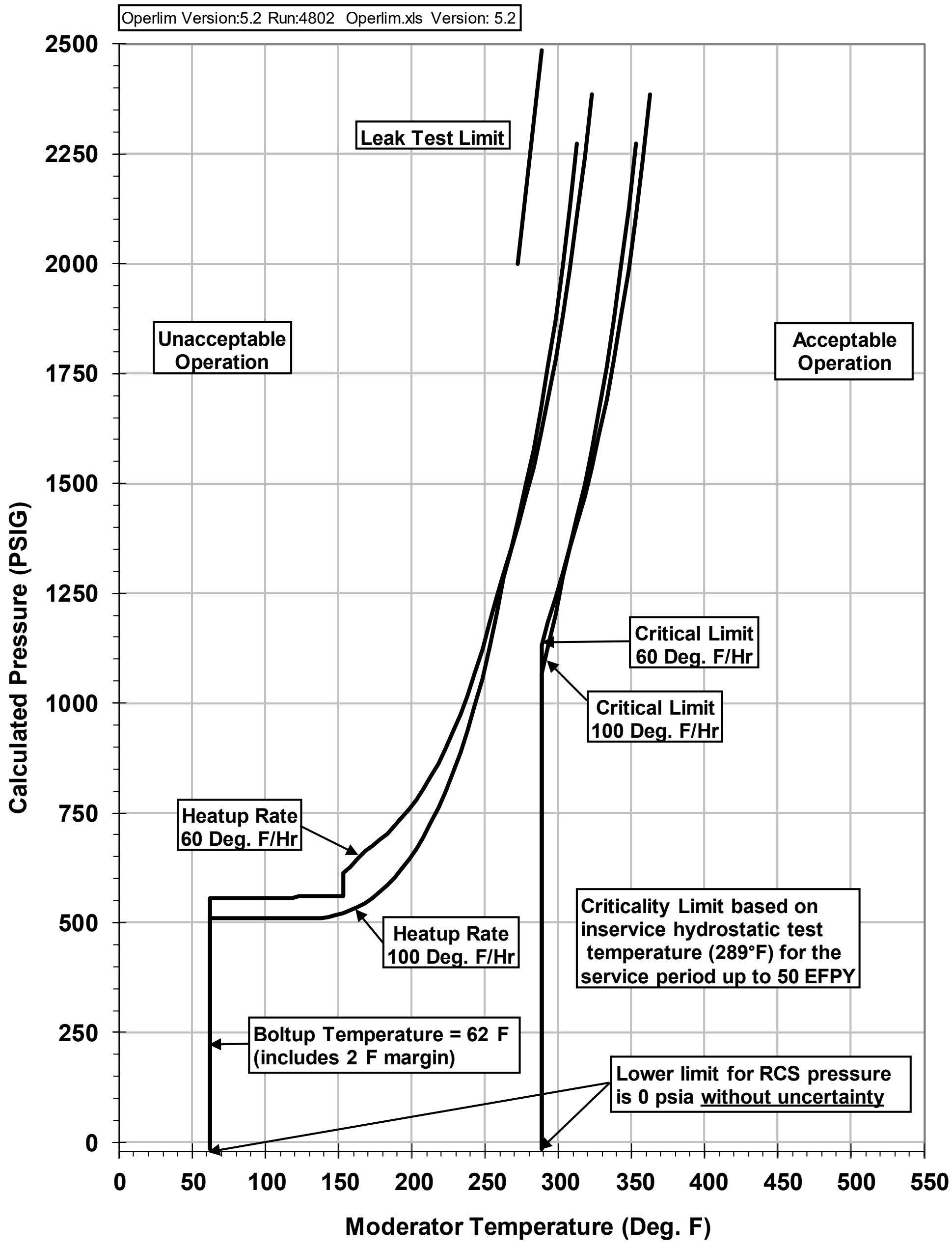

Figure 3-1 Salem Unit 2 Reactor Coolant System Heatup Limitations Applicable for Heatup Rates up to 100°F/HR for the Service Period up to 50 EFPY (with uncertainties for instrumentation errors)

LIMITING MATERIAL AND ART VALUES AT 50 EFPY:
1/4T ART: 209°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)
3/4T ART: 150°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)

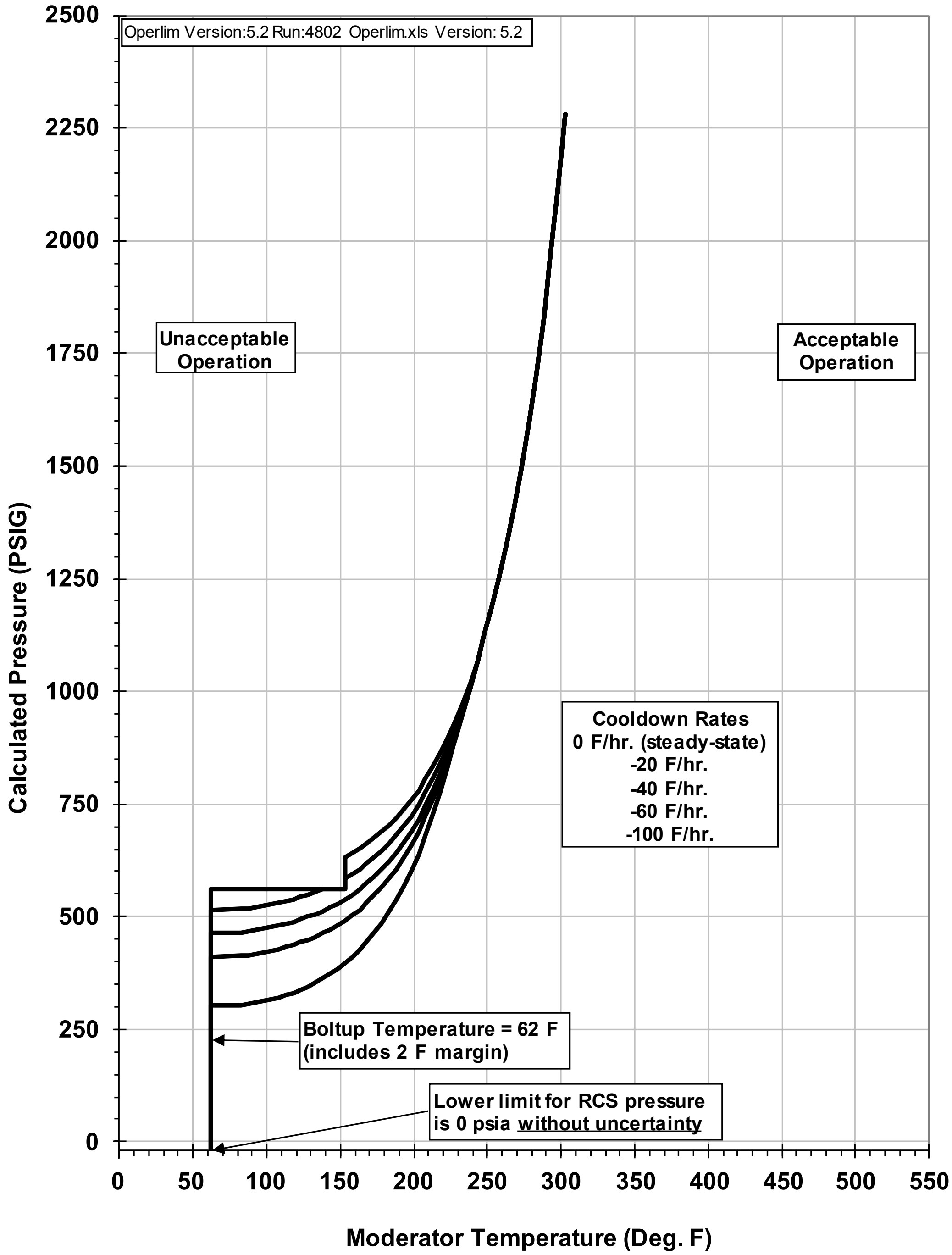

Figure 3-2 Salem Unit 2 Reactor Coolant System Cooldown Limitations Applicable for Cooldown Rates up to 100°F/HR for the Service Period up to 50 EFPY (with instrumentation errors)

4.0 Discussion

4.1 Salem Unit 2 Surveillance Program

Appendix A to this PTLR contains the Salem Unit 2 Surveillance Program.

4.2 General Overview of Development of P-T Limit Curves

The analyses that support the generation of the Heatup and Cooldown Pressure-Temperature (P-T) limit curves presented in this PTLR are contained in References [1] and [2]. Reference [2] is a verification of the P-T limit curves originally developed in [1] for license renewal.

The P-T limit curves were generated through End of License Extension (EOLE), i.e., 60 years of operation, using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G [3]. The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [4]. The heatup and cooldown P-T limit curves utilize the limiting Adjusted Reference Temperature (ART) values ($RT_{NDT(U)}$) plus ΔRT_{NDT} plus margins for uncertainties) at the quarter thickness (1/4T) and three-quarter thickness (3/4T) locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface for Salem Unit 2. The ART values for 50 effective full-power years (EFPY) were calculated using Regulatory Guide 1.99, Revision 2 [5].

The Salem Unit 2 P-T limit curves were generated with instrumentation errors, and the reactor vessel flange requirements of 10 CFR 50, Appendix G [13].

4.3 Fluence Calculations

To evaluate fast neutron exposure for the RPV, a three-dimensional discrete ordinates code, RAPTOR-M3G [7] was used, along with the BUGLE-96 cross-section library [14]. Radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. Projections of future operation were based on the spatial power distributions and reactor operating conditions of Salem Unit 2 Cycles 20 and 23.

The neutron fluence was calculated as follows. Discrete ordinates (S_N) transport analyses were performed to determine the neutron radiation environment within the RPV. In these analyses, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. The dosimetry analysis documented in Appendix C of [2] shows that the ±20% (1σ) acceptance criteria specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [6], is met, based on the measurement-to-calculation (M/C) comparison results for the in-vessel surveillance capsules withdrawn and analyzed to-date. These validated calculations form the basis for providing projections of the neutron exposure of the RPV through EOLE.

All of the calculations were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET" [7]. The neutron transport evaluation methodology is based on the guidance of Regulatory Guide 1.190.

The uncertainty associated with the calculated neutron exposure of the Salem Unit 2 RPV is based on the recommended approach provided in Regulatory Guide 1.190. The qualification of the methodology used in the plant-specific neutron exposure evaluation is carried out in the following four stages:

Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator (NUREG/CR-6454, "Pool Critical Assembly Pressure Vessel Facility Benchmark" [8]) at the Oak Ridge National Laboratory (ORNL) and the VENUS-1 experiment.

Comparison of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment (NUREG/CR-6453, "H.B. Robinson 2 Pressure Vessel Benchmark" [9]).

An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments (WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET" [7]).

Comparison of calculations with all available dosimetry results from the RPV measurement programs carried out at Salem Unit 2 (Appendix C of [2]).

Table 4-1 (re-produced from Table 6-1 of [2]) summarizes the fluence projections at 50 EFPY for the Vessel Surface, 1/4T, and 3/4T locations for the Salem Unit 2 RPV. These fluence values were used to calculate the end of life extension (EOLE) ART values for the Salem Unit 2 RPV.

Table 4-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T and 3/4T Locations for the Salem Unit 2 Reactor Vessel Materials at 50 EFPY

Material	Surface fluence[a] (x 10^{19} n/cm^2, E> 1.0 MeV)	Surface FF[c]	1/4T fluence[b] (x 10^{19} n/cm^2, E> 1.0 MeV)	1/4T FF[c]	3/4T fluence[b] (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF[c]
Upper Shell Plates	0.0244	0.194	0.0145	0.141	0.00517	0.069
Intermediate Shell Plates	2.05	1.196	1.22	1.056	0.434	0.768
Lower Shell Plates	2.01	1.190	1.20	1.050	0.426	0.763
Intermediate to Upper Shell Circumferential Weld Seam 8-442	0.028	0.211	0.0167	0.154	0.00593	0.076
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.98	1.186	1.18	1.046	0.419	0.759
Upper Shell Longitudinal Weld Seams 1-442A & C	0.0183	0.163	0.0109	0.116	0.00387	0.055
Upper Shell Longitudinal Weld Seam 1-442B	0.0101	0.110	0.0060	0.077	0.00214	0.035
Intermediate Shell Longitudinal Weld Seams 2-442 B & C	1.47	1.107	0.876	0.963	0.311	0.680
Intermediate Shell Longitudinal Weld Seam 2-442A	0.741	0.916	0.442	0.773	0.157	0.513
Lower Shell Longitudinal Weld Seams 3-442 A & C	1.45	1.103	0.864	0.959	0.307	0.676
Lower Shell Longitudinal Weld Seams 3-442B	0.738	0.915	0.440	0.772	0.156	0.512

Note(s):

(a) Fluence values are documented in Table 2-2 of [2].

(b) The 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (8.625 inches) and equation $f = f_{surf} * e^{-0.24 (x)}$ from Regulatory Guide 1.99, Revision 2, where x = the depth into the vessel wall (inches).

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log (f))}$.

4.4 Fracture Toughness Properties

P-T limit curve development requirements are specified in 10 CFR 50, Appendix G [13], which also defines the beltline region of the RPV. Any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves, where these additional materials that exceed this fluence threshold are referred to as the "extended beltline" materials and are evaluated to ensure that the applicable neutron embrittlement effects are considered.

For Salem Unit 2, the extended beltline materials include upper shell plates, upper shell longitudinal welds, and the upper to intermediate shell girth weld. However, the fluence for both inlet/outlet nozzle to upper shell welds are less than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at 50 EFPY, therefore, the nozzle forgings and associated welds to the upper shell do not need to be considered in the extended beltline.

Note: for RPV welds, the terms "girth" and "circumferential" are used interchangeably and for the purposes of this PTLR and its supporting documentation, these welds shall be referred to as circumferential welds. Similarly, the terms "axial" and "longitudinal" are used interchangeably; herein, these welds are referred to as longitudinal welds.

Although the RPV nozzles are not a part of the extended beltline, per NRC RIS 2014-11 [11], the nozzle materials must be evaluated for their potential effect on P-T limit curves due to the higher stresses in the nozzle corner region. The effects of these higher stresses are addressed in [2], which determines that the Salem Unit 2 beltline P-T limit curves bound the inlet and outlet nozzle P-T limit curves.

Table 3-1 of [2] contains the summary of the best-estimate copper and nickel contents in units of weight percent (wt. %), as well as initial RT_{NDT} values, for the Salem Unit 2 reactor vessel beltline and extended beltline materials. Table 3-2 of [2] contains the summary of the best-estimate copper and nickel contents in units of weight percent (wt. %), as well as initial RT_{NDT} values, for the Salem Unit 2 reactor vessel inlet and outlet nozzle materials. Table 3-1 of [1] lists the initial RT_{NDT} values for the Replacement Reactor Vessel Closure Head (RRVCH) and RPV flange materials.

4.5 Use of Surveillance Data

Position 2.1 of RG 1.99, Rev. 2 [5] requires data from the plant-specific surveillance program to calculate chemistry factors. Table 4-1 of [2] lists the capsule surveillance data for Salem Unit 2 to date, and Table 4-2 of [2] lists data from a surveillance program at a sister plant, Diablo Canyon Unit 2, which includes a Salem Unit 2 RPV extended beltline material that should also be considered when calculating Position 2.1 chemistry factors. Appendix C of this PTLR reproduces the credibility re-evaluation of the surveillance data as contained in Appendix A of [2].

4.6 Chemistry Factors

The chemistry factors (CFs) were calculated using Regulatory Guide 1.99, Revision 2 [5], Positions 1.1 and 2.1. Since Salem Unit 2 has a plant-specific surveillance program, Position 2.1 chemistry factor calculations were performed using the method described in Regulatory Guide 1.99, Revision 2 [5]. Table 4-2 of this PTLR (re-produced from Table 4-1 of [2]) summarizes the surveillance data available for the Salem Unit 2 plate and weld materials that were used in the calculation of the Position 2.1 chemistry factor values. The Position 2.1 chemistry factor calculations are presented in Table 4-3 of this PTLR (re-produced from Table 5-1 of [2]) for the Salem Unit 2 surveillance materials.

Salem Unit 2 considered surveillance data from a sister plant, Diablo Canyon Unit 2, for weld heat # 21935/12008. Adjustment of the ΔRT_{NDT} values were required per Regulatory Guide 1.99, Revision 2 [5] due to chemistry and irradiation temperature differences between the surveillance welds and Salem Unit 2 RPV welds. Position 2.1 chemistry factor calculations using the Diablo Canyon Unit 2 surveillance data for weld heat # 21935/12008 were performed using the method described in Regulatory Guide 1.99, Revision 2 [5]. Table 4-4 of this PTLR (re-produced from Table 4-2 of [2]) summarizes the surveillance data available for the Diablo Canyon Unit 2 weld materials that were used in the calculation of the Position 2.1 chemistry factor values. The Position 2.1 chemistry factor calculations are presented in Table 4-5 of this PTLR (re-produced from Table 5-2 of [2]) for the Diablo Canyon Unit 2 surveillance materials.

Table 4-2 Salem Unit 2 Surveillance Capsule Data(a)

Material	Capsule	Fluence(b) (n/cm^2, E > 1.0 MeV)	Measured ΔRT_{NDT} (°F)	Irradiation Temperature (°F)
Intermediate Shell Plate B4712-2 (Longitudinal)	T	0.273E+19	61.66	545
	U	0.581E+19	66.54	543
	X	1.13E+19	93.82	541
	Y	1.83E+19	105.69	540
Intermediate Shell Plate B4712-2 (Transverse)	T	0.273E+19	74.83	545
	U	0.581E+19	98.26	543
	X	1.13E+19	125.15	541
	Y	1.83E+19	129.33	540
Salem Unit 2 Surveillance Weld Material(c) (Heat # 13253)	T	0.273E+19	153.17	545
	U	0.581E+19	185.94	543
	X	1.13E+19	195.43	541
	Y	1.83E+19	200.90	540

Note(s):

(a) Information extracted from WCAP-15692 {refer to Ref. 15 of [1]}, unless otherwise noted.

(b) The fluence values were taken from Table 2-4 of [2].

(c) Note that the Salem Unit 2 surveillance weld metal Heat # 13253 is only representative of the beltline welds, not identical. Therefore, the surveillance weld data was not used in the calculations documented in [1] and [2].

Table 4-3 Calculation of Salem Unit 2 Chemistry Factors Using Surveillance Capsule Data[(a)]

Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Measured ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate B4712-2 (Longitudinal)	T	0.273	0.646	61.66	39.84	0.418
	U	0.581	0.848	66.54	56.43	0.719
	X	1.13	1.034	93.82	97.02	1.069
	Y	1.83	1.199	105.69	123.21	1.359
Intermediate Shell Plate B4712-2 (Transverse)	T	0.273	0.646	74.83	48.35	0.418
	U	0.581	0.848	98.26	83.33	0.719
	X	1.13	1.034	125.15	129.42	1.069
	Y	1.83	1.199	129.33	150.76	1.359
	SUM:				728.36	7.130
	$CF_{B4712-2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (728.36) \div (7.130) = 102.2°F$					
Salem Unit 2 Surveillance Weld Material (Heat # 13253)	T	0.273	0.646	153.17	98.97	0.418
	U	0.581	0.848	185.94	157.68	0.719
	X	1.13	1.034	195.43	202.10	1.069
	Y	1.83	1.199	200.90	234.20	1.359
				SUM:	692.95	3.565
	$CF_{Surv. Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (692.95) \div (3.565) = 194.4°F$					

Note(s):

(a) Data taken from Table 4-1 of [2], unless otherwise noted.

(b) FF = fluence factor = $f^{(0.28 - 0.101*\log(f))}$.

Table 4-4 Sister-Plant (Diablo Canyon Unit 2) Surveillance Program Results

Material	Capsule	Withdrawal		Capsule Fluence ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	Measured ΔRT_{NDT} (°F)	Irradiation Temperature (°F)
		Cycle	EFPY			
Diablo Canyon Unit 2 Surveillance Weld Material[(a)] (Heat # 21935/12008)	U	1	1.02	0.330	173.0	545
	X	3	3.16	0.906	203.2	545
	Y	6	7.08	1.53	211.4	545
	V	9	11.49	2.38	224.5	545

Note(s):

(a) The Diablo Canyon Unit 2 capsule data were taken from WCAP-17315-NP (Ref. 15 of [2]). EFPY values are found in WCAP-15423 (Ref. 12 of [2]).

Table 4-5 Calculation of Chemistry Factors Using Diablo Canyon Unit 2 Surveillance Capsule Data[(a)]

Material	Capsule	Capsule Fluence ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(b)]	Measured ΔRT_{NDT}[(b)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Diablo Canyon Unit 2 Surveillance Weld Material (Heat # 21935/12008)	U	0.330	0.695	176.0 (173.0)	122.32	0.483
	X	0.906	0.972	206.2 (203.2)	200.49	0.945
	Y	1.53	1.118	214.4 (211.4)	239.62	1.249
	V	2.38	1.234	227.5 (224.5)	280.70	1.522
				SUM:	843.14	4.200
	$CF_{Surv. Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (843.14) \div (4.200) = 200.7°F$					

Note(s):

(a) Fluence and ΔRT_{NDT} taken from Table 4-2 of [2].
(b) FF = fluence factor = $f^{(0.28 - 0.101*\log(f))}$
(c) The surveillance weld ΔRT_{NDT} values have been adjusted for irradiation temperature as follows:
Adjusted $\Delta RT_{NDT} = \Delta RT_{NDT,\ Measured}$ + Temp. Adjustment
The temperature adjustments are based on a Salem Unit 2 reactor vessel temperature of 542°F. The Diablo Canyon Unit 2 capsule irradiation temperatures are provided in Table 4-2 of [2]. The measured (unadjusted) ΔRT_{NDT} values are shown in parenthesis. The temperature adjustment is therefore 3°F (545°F - 542°F). Note, CF ratio procedure ($CF_{Salem\ 2\ vessel\ weld} / CF_{Diablo\ Canyon\ 2\ surv.\ weld}$) results in a ratio of less than 1.0; therefore, the chemistry adjustment can conservatively be neglected.

4.7 Criteria for Allowable Pressure-Temperature Relationships

The approach to the development of the Salem Unit 2 P-T limit curves, including their equations and figures from the ASME Code, Section XI, Appendix G is contained in [1]. The P-T limit curve methodology is the same as that described in WCAP-14040-A [4]. The RPV metal temperature at the crack tip of a postulated flaw is determined based on the methodology contained in Section 2.6.1 of WCAP-14040-A.

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic}$$

where,

K_{Im} =	stress intensity factor caused by membrane (pressure) stress
K_{It}=	stress intensity factor caused by the thermal gradients
K_{Ic}=	reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}
C=	2.0 for Level A and Level B service limits
C=	1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{Ic} is determined by using the metal temperature at the tip of a postulated flaw (1/4T) and the RT_{NDT}, in the reference fracture toughness curve equation, Equation 2.5-2 from [4]. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw were computed, the pressure stress intensity factors, K_{Im}, were then obtained, and from these, the allowable pressures were calculated.

4.8 Closure Head / Vessel Flange Requirements

Regarding RPV closure head flange and RPV flange regions, 10 CFR Part 50, Appendix G [13] states that the metal temperature of the closure head regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which was calculated to be 621 psig [1]. The RPV closure head and RPV flange initial RT_{NDT} values (shown as $RT_{NDT(U)}$) are provided in [1]. The limiting unirradiated RT_{NDT} of 12°F (rounded up to 15°F) is associated with the vessel flange of the Salem Unit 2 RPV, so the minimum allowable temperature of this region is 153°F for normal operation at pressures greater than 560 psig <u>with</u> margins for instrument uncertainties.

4.9 Boltup Temperature Requirements

Boltup temperature requirements are established in Appendix G to 10 CFR 50 [13]. Per [4], the minimum boltup temperature should be 60°F <u>or</u> the limiting unirradiated RT_{NDT} of the RPV closure flange region, whichever is higher. The minimum boltup temperature for the Salem Unit 2 RPV is 60°F <u>without</u> margins for instrument uncertainties and 62°F <u>with</u> margins for instrument uncertainties.

4.10 Calculation of ART

The formulas and methodology for calculating the end of life extension (EOLE) ART values, which are based on fluence calculated at 50 EFPY, are provided in [2]. Salem Unit 2 had generated P-T Curves as part of license renewal [1], but were not submitted for approval at the time. In order to support the implementation of the 50 EFPY P-T limit curves in Figure 3-1 and 3-2 of this PTLR, the applicability of the P-T limit curves was evaluated by comparing the 50 EFPY limiting 1/4T and 3/4T ART values contained in Tables 6-2 and 6-3 of [2] with those used developed in [1].

Table 4-6 contains the limiting ART values for the Salem Unit 2 RPV at 50 EFPY (partially reproduced from Table 7-1 of [2]).

Table 4-6
Limiting ART Values for Salem Unit 2 at 50 EFPY

	Verification Analysis[a]		**P-T Limit Curves Analysis**[b]	
	Limiting ART Values	**Limiting Material**	**Limiting ART Value**)	**Limiting Material**
1/4T Location	191.6[c]	Intermediate Shell Longitudinal Weld Seams 2-442 B & C (Position 1.1)	209	Lower Shell Longitudinal Weld Seam 3-442 A & C (Position 1.1)
3/4T Location	138.1[c]	Intermediate Shell Longitudinal Weld Seams 2-442 B & C (Position 1.1)	150	Lower Shell Longitudinal Weld Seam 3-442 A & C (Position 1.1)

Note(s):

(a) Values are the limiting values from Tables 6-2 and 6-3 of [2].

(b) Values are the limiting values as listed in Table 6-1 of [1]. The P-T Curves developed in [1] are based on more conservative ART values computed in [2], therefore, are the governing curves for this PTLR.

(c) Note, the ART value calculated for Lower Shell Longitudinal Weld Seams 3-442 A & C have higher ART values at both the 1/4T and 3/4T locations when a Position 1.1 CF is used [1]. Because credible surveillance data exists from the Diablo Canyon Unit 2 (sister plant) surveillance program, the lower 1/4T and 3/4T ART values, calculated with a Position 2.1 CF, as documented in [2], supersede those values. Therefore, the Limiting ART values to use in [2] for the P-T Curve applicability are 191.6°F and 138.1°F for 1/4T and 3/4T locations, respectively.

The P-T limit curves developed in [1] remain valid since they are based on the bounding ART values listed above in Table 4-6.

4.11 Pressurized Thermal Shock (PTS)

10 CFR 50.61 establishes screening criteria on pressurized water reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld

seams for plant operation to the end of plant license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

These accepted methods were used with the surface fluence values to calculate the following RT_{PTS} values for the Salem Unit 2 RPV materials. The results of the RT_{PTS} calculations are presented in Table B-1 of [2] for Salem Unit 2.

All of the reactor vessel materials for Salem Unit 2 are below the RT_{PTS} screening criteria values of 270°F for plates, forgings, and longitudinal welds, and 300°F for circumferentially oriented welds (per 10 CFR 50.61) at 50 EFPY.

4.12 Heatup and Cooldown P-T Limit Curves

For P-T limit curve development, the limiting ART values from Table 4-4 were used, i.e., 209°F and 150°F for the limiting 1/4T and 3/4T locations, respectively.

The Salem Unit 2 P-T limit curves are presented in Section 3 of this PTLR, which incorporate all necessary limits for ensuring prevention of non-ductile failure for the Salem Unit 2 RPV with the flange requirements and with instrumentation uncertainties. The data points used to construct the P-T limit curves are shown in Tables B-1, B-2, and B-3 of Appendix B to this PTLR. Vacuum refill limits for the RCS are depicted on the P-T Curves by showing a minimum pressure of 0 psia.

4.13 LTOP Enable Temperature

ASME Code Case N-641 [15] includes a provision for calculating P-T relationships and low temperature overpressure protection (LTOP) system effective temperatures, T_e, and allowable pressures. The enable temperature presented in [1] is based on the same ART values as the P-T limit curves. Since the P-T limit curves developed in [1] remain valid, the enable temperature reported in [1] also remains valid. Thus, the minimum required enable temperature without margins for instrument uncertainty is a coolant temperature equal to 274°F through EOLE. With margins for instrument uncertainty (+18°F), the minimum required enable temperature is a coolant temperature equal to 292°F. This analyzed enable temperature is lower than the current 32 EPFY setpoint of 312°F (more conservative), which will remain as the enable temperature setpoint as stated in Section 3.0 of this PTLR. Salem Unit 2 may revise the enable temperature setpoint to a value down to 292°F following an approved design change.

4.14 Pressurizer Overpressurization Protection System (POPS) Analysis

The LTOPS is also known as the POPS at Salem Unit 2 and consists of the two Power Operated Relief Valves (PORVs) with reduced relief settings. These setpoints were selected following the NRC-approved methodology in WCAP-14040-A, Rev. 4 [4] such that the peak pressure during the design basis Mass Injection (MI) and Heat Injection (HI) transients will not exceed the isothermal 10CFR50 Appendix G [13] P-T limits.

The analyses that support the generation of the POPS PORV setpoint presented in this PTLR are contained in References [16] and [17]. Reference [16] is a verification of the POPS setpoint originally developed in [17] for license renewal.

The final maximum allowable PORV setpoint is determined such that it bounds both the MI and HI transient maximum allowable PORV setpoints.

The analyzed Salem Unit 2 POPS PORV setting is 434 psig [16], which is higher than the current 32 EPFY setpoint of 375 psig (more conservative) as stated in Section 3.0 of this PTLR. Salem Unit 2 may revise the POPS PORV setpoint to a value up to 434 psig following an approved design change.

5.0 References

1. PSEG VTD 901764 (001), Westinghouse WCAP-16982-NP, Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity (Non-Proprietary Class 3), Rev. 3, December 2020
2. PSEG VTD 904297 (001), Westinghouse WCAP-18571-NP, Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation (Non-Proprietary Class 3), Rev. 2, July 2022
3. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."
4. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004
5. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
6. Regulatory Guide 1.190, Revision 0, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.
7. Westinghouse Report WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018.

8. ORNL Report ORNL/TM-13205, "Pool Critical Assembly Pressure Vessel Facility Benchmark," (NUREG/CR-6454), July 1997.
9. ORNL Report ORNL/TM-13204, "H. B. Robinson-2 Pressure Vessel Benchmark," (NUREG/CR 6453), October 1997 (published February 1998).
10. 10CFR50.59, "Changes, tests, and experiments".
11. NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014.
12. ASME Boiler and Pressure Vessel Code Section III.
13. Appendix G to Part 50—Fracture Toughness Requirements.
14. RSICC Data Library Collection DLC-185, "BUGLE-96: Coupled 47 Neutron, 20 GammC-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," July 1999.
15. ASME Code Case N-641.
16. PSEG VTD 904298 (001), Westinghouse Letter Report LTR-SCS-20-52-P, Rev. 1, "Salem Unit 2 Low Temperature Overpressure Protection System (LTOPS) / Pressurizer Overpressure Protection System (POPS) Analysis," July 26, 2021.
17. PSEG VTD 904298 (003), Westinghouse Letter PNG-LTP-TR-AA-000001-P, Rev. 1, "Transmittal of PSE-09-18 – "Low Temperature Overpressure Protection System (LTOPS) Setpoint Analysis for Salem Unit 1 and 2" to Support the Salem Unit 2 LAR.

Appendix A

SALEM UNIT 2 REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM

The Salem Unit 2 Surveillance Program is described in UFSAR Section 5.2.4.4.

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements [A-1], four (4) surveillance capsules have been removed from the Salem Unit 2 reactor vessel in accordance with the following schedule.

Capsule T Removed in 1983
WCAP-10492, Analysis of Capsule T from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program, March 1984

Capsule U Removed in 1986
WCAP-11554, Analysis of Capsule U from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program, September 1987

Capsule X Removed in 1991
WCAP-13366, Analysis of Capsule X from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program, June 1992

Capsule Y Removed in 2000
WCAP-15692, Analysis of Capsule Y from the Public Service Electric and Gas Co. Salem Unit 2 Reactor Vessel Radiation Surveillance Program, August 2001

Capsule S Scheduled to be removed at 32 EFPY, tentatively May 2023

Capsule V Standby
Capsule W Standby
Capsule Z Standby

APPENDIX A REFERENCES:

A-1 U.S. Code of Federal Regulations, Title 10, Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” October 2, 2020.

Appendix B

TABULATED VALUES FOR SALEM UNIT 2 P-T CURVES

Table B-1 (Re-produced from Table 6-2 of [1])
50 EFPY Heatup Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
62	Note (a)	289	Note (a)	62	Note (a)	289	Note (a)
62	555	289	555	62	510	289	510
83	555	289	555	83	510	289	511
88	555	289	556	88	510	289	511
93	555	289	558	93	510	289	513
98	555	289	560	98	510	289	514
103	555	289	560	103	510	289	516
108	555	289	560	108	510	289	518
113	555	289	560	113	510	289	521
118	556	289	560	118	510	289	523
123	560	289	560	123	510	289	528
128	560	289	560	128	510	289	531
133	560	289	560	133	510	289	536
138	560	289	560	138	511	289	541
143	560	289	560	143	513	289	546
148	560	289	560	148	516	289	552
153	560	289	613	153	521	289	557
153	560	289	627	153	521	289	560
153	613	289	644	153	521	289	560
158	627	289	663	158	528	289	560
163	644	289	684	163	536	289	560
168	662	289	707	168	546	289	570
173	675	289	733	173	557	289	585

**Table B-1 (Re-produced from Table 6-2 of [1])
50 EFPY Heatup Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]**

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
178	688	289	762	178	570	289	603
183	703	289	794	183	585	289	622
188	720	289	829	188	603	289	644
193	738	289	868	193	622	289	669
198	758	289	906	198	644	289	696
203	781	289	927	203	669	289	727
208	805	289	950	208	696	289	761
213	833	289	976	213	727	289	798
218	863	289	1003	218	761	289	840
223	896	289	1034	223	798	289	886
228	933	289	1068	228	840	289	938
233	973	289	1105	233	886	289	994
238	1018	289	1132	238	938	289	1067
243	1068	293	1184	243	994	293	1126
248	1123	298	1241	248	1057	298	1203
253	1184	303	1296	253	1126	303	1287
258	1241	308	1356	258	1203	308	1354
263	1296	313	1423	263	1287	313	1409
268	1356	318	1496	268	1354	318	1470
273	1423	323	1577	273	1409	323	1537
278	1496	328	1667	278	1470	328	1611
283	1577	333	1766	283	1537	333	1692

Table B-1 (Re-produced from Table 6-2 of [1])
50 EFPY Heatup Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
288	1667	338	1875	288	1611	338	1782
293	1766	343	1995	293	1692	343	1880
298	1875	348	2128	298	1782	348	1989
303	1995	353	2274	303	1880	353	2109
308	2128			308	1989	358	2241
313	2274			313	2109	363	2386
				318	2241		
				323	2386		

Note:

(a) The lower limit for reactor coolant system pressure is 0 psia without uncertainty.

Table B-2 (Re-produced from Table 6-3 of [1])
50 EFPY Cooldown Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)
62	Note (a)	62	Note (a)	62	Note (a)	62	Note (a)	62	Note (a)
62	560	62	515	62	464	62	411	62	302
83	560	83	517	83	466	83	413	83	304
88	560	88	519	88	468	88	415	88	306
93	560	93	522	93	470	93	418	93	309
98	560	98	525	98	473	98	421	98	313
103	560	103	528	103	476	103	424	103	316
108	560	108	531	108	480	108	428	108	321
113	560	113	535	113	484	113	432	113	326
118	560	118	539	118	489	118	437	118	331
123	560	123	544	123	494	123	442	123	338
128	560	128	549	128	499	128	449	128	345
133	560	133	555	133	506	133	455	133	353
138	560	138	560	138	513	138	463	138	362
143	560	143	560	143	520	143	471	143	372
148	560	148	560	148	529	148	481	148	383
153	560	153	560	153	539	153	491	153	396
153	560	153	560	158	549	158	503	158	410
153	632	153	586	163	561	163	516	163	426
158	641	158	595	168	574	168	530	168	444
163	651	163	606	173	589	173	546	173	463
168	662	168	618	178	605	178	564	178	485
173	675	173	632	183	623	183	584	183	510
178	688	178	646	188	643	188	606	188	537
183	703	183	663	193	665	193	630	193	567

Table B-2 (Re-produced from Table 6-3 of [1])
50 EFPY Cooldown Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)
188	720	188	681	198	690	198	658	198	601
193	738	193	701	203	717	203	688	203	638
198	758	198	723	208	747	208	721	208	680
203	781	203	748	213	780	213	758	213	726
208	805	208	775	218	817	218	799	218	777
213	833	213	805	223	858	223	844	223	834
218	863	218	838	228	903	228	895	228	895
223	896	223	875	233	953	233	950	233	950
228	933	228	916	238	1008	238	1008	238	1008
233	973	233	961	243	1066	243	1066	243	1066
238	1018	238	1011	248	1123	248	1123	248	1123
243	1068	243	1066	253	1184	253	1184	253	1184
248	1123	248	1123	258	1251	258	1251	258	1251
253	1184	253	1184	263	1325	263	1325	263	1325
258	1251	258	1251	268	1407	268	1407	268	1407
263	1325	263	1325	273	1497	273	1497	273	1497
268	1407	268	1407	278	1597	278	1597	278	1597
273	1497	273	1497	283	1708	283	1708	283	1708
278	1597	278	1597	288	1830	288	1830	288	1830
283	1708	283	1708	293	1965	293	1965	293	1965
288	1830	288	1830	298	2115	298	2115	298	2115
293	1965	293	1965	303	2280	303	2280	303	2280
298	2115	298	2115						
303	2280	303	2280						

Note:

(a) The lower limit for reactor coolant system pressure is 0 psia without uncertainty.

Table B-3 (Re-produced from Table 6-2 of [1])
Salem Unit 2 50 EFPY Inservice Leak Test Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (with K_{Ic}, and Margins for Temperature (18°F) and Pressure (61 psi) Instrumentation Errors)

Temperature (°F)	Pressure (psig)
272	2000
289	2485

Appendix C

SALEM UNIT 2 SURVEILLANCE DATA CREDIBILITY EVALUATION

Regulatory Guide 1.99, Revision 2 [Ref. C-1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Positions 2.1 and 2.2 of Regulatory Guide 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date, there have been four surveillance capsules removed and tested from the Salem Unit 2 reactor vessel. In addition, surveillance data from a sister plant is utilized herein to perform neutron radiation embrittlement calculations.

Table C-1 reviews the five criteria of Regulatory Guide 1.99, Revision 2. The following subsections evaluate each of these five criteria for Salem Unit 2 in order to determine the credibility of the surveillance data for use in neutron radiation embrittlement calculations.

Table C-1 Identification of Credibility Criteria Affected by Fluence Change

Criterion No.	Description
1	Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.
2	Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.
3	When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. C-4].
4	The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.
5	The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR 50, "Fracture Toughness Requirements," [Ref. C-2] as follows:

> *... the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

The Salem Unit 2 reactor vessel consists of the following beltline region materials, which likely would have been considered at the time the surveillance program was designed and licensed:

- Intermediate Shell Plates B4712-1, B4712-2, and B4712-3
- Lower Shell Plates B4713-1, B4713-2, and B4713-3
- Intermediate to Lower Shell Circumferential Weld Seam 9-442
  (Heat # 90099, Linde 0091 flux, Lot # 3977)
- Intermediate Shell Longitudinal Weld Seams 2-442 A, 2-442 B, and 2-442 C
  (Heat # 13253/20291, Linde 1092 flux, Lot # 3833)
- Lower Shell Longitudinal Weld Seams 3-442 A, 3-442 B, and 3-442 C
  (Heat # 21935/12008, Linde 1092 flux, Lot # 3889)

Intermediate Shell Plate B4712-2 has the highest initial RT_{NDT} of the beltline plates and was selected for the surveillance program. The surveillance weld was fabricated with weld wire Heat # 13253 and Linde 1092 flux, Lot # 3833/3774 utilizing the same fabrication practice as that used to fabricate the actual vessel beltline welds. This weld was made of one of the heats and with the same type flux as was used in the intermediate shell longitudinal weld seams.
Thus, the Salem Unit 2 surveillance program meets the intent of this criterion.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

The surveillance capsule analysis report, WCAP-15692 [Ref. C-5], was reviewed and it was determined that this criterion is met, consistent with the analysis in these documents.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. C-4].

The functional form of the least-squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the plates.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 [Ref. C-3]. At this meeting, the NRC presented five cases. Of the five cases, Case 1 ("Surveillance Data Available from Plant but No Other Source") most closely represents the situation for the Salem Unit 2 surveillance plate and weld materials. Case 5 ("Surveillance Data from Other Sources Only") most closely represents the situation for the Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442 A, 3-442 B, and 3-442 C (Heat # 21935/12008) weld material.

Evaluation of the Salem Unit 2 Data Only (Case 1)

Following the NRC Case 1 guidelines, the Salem Unit 2 surveillance plates and weld metal (Heat # 13253) will be evaluated using the Salem Unit 2 data. Table C-2 provides the calculation of the interim CF for Salem Unit 2. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2 Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values. In addition, only Salem Unit 2 data are being considered; therefore, no temperature adjustment is required.

Table C-2 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using Salem Unit 2 Surveillance Capsule Data Only

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(a)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate B4712-2 (longitudinal)	T	0.273	0.646	61.66	39.84	0.418
	U	0.581	0.848	66.54	56.43	0.719
	X	1.13	1.034	93.82	97.02	1.069
	Y	1.83	1.166	105.69	123.21	1.359
Intermediate Shell Plate B4712-2 (transverse)	T	0.273	0.646	74.83	48.35	0.418
	U	0.581	0.848	98.26	83.33	0.719
	X	1.13	1.034	125.15	129.42	1.069
	Y	1.83	1.166	129.33	150.76	1.359
	SUM:				728.36	7.130
	$CF_{B4712-2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (728.36) \div (7.130) = 102.2°F$					
Salem Unit 2 surveillance weld material (Heat # 13253)	T	0.273	0.646	153.17	98.97	0.418
	U	0.581	0.848	185.94	157.68	0.719
	X	1.13	1.034	195.43	202.10	1.069
	Y	1.83	1.166	200.90	234.20	1.359
	SUM:				692.95	3.565
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (692.95) \div (3.565) = 194.4°F$					

Notes:

(a) Fluence and measured ΔRT_{NDT} taken from Table 4-1 of [2].

(b) FF = fluence factor = $f^{(0.28 - 0.10 * \log(f))}$.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Guide 1.99, Revision 2 Position 2.1 is presented in Table C-3.

Table C-3 Surveillance Capsule Data Scatter about the Best-Fit Line Using Only Salem Unit 2 Surveillance Data

Material	Capsule	CF[(a)] (Slope$_{best-fit}$) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(c)]	Measured ΔRT_{NDT}[(b)] (°F)	Predicted ΔRT_{NDT}[(d)] (°F)	Scatter ΔRT_{NDT}[(e)] (°F)	< 17°F (Base Metal) < 28°F (Weld)
Intermediate Shell Plate B4712-2 (longitudinal)	T	102.2	0.273	0.646	61.66	66.0	4.4	Yes
	U	102.2	0.581	0.848	66.54	86.6	20.1	No
	X	102.2	1.13	1.034	93.82	105.6	11.8	Yes
	Y	102.2	1.83	1.166	105.69	119.1	13.4	Yes
Intermediate Shell Plate B4712-2 (transverse)	T	102.2	0.273	0.646	74.83	66.0	8.8	Yes
	U	102.2	0.581	0.848	98.26	86.6	11.6	Yes
	X	102.2	1.13	1.034	125.15	105.6	19.5	No
	Y	102.2	1.83	1.166	129.33	119.1	10.2	Yes
Salem Unit 2 surveillance weld material (Heat # 13253)	T	194.4	0.273	0.646	153.17	125.6	27.6	Yes
	U	194.4	0.581	0.848	185.94	164.8	21.1	Yes
	X	194.4	1.13	1.034	195.43	201.0	5.6	Yes
	Y	194.4	1.83	1.166	200.90	226.6	25.7	Yes

Notes:

(a) CFs calculated in Table C-2.
(b) Fluence and measured ΔRT_{NDT} values are taken from Table 5-1 of [2].
(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(d) Predicted ΔRT_{NDT} = CF x FF.
(e) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

From a statistical point of view, +/- 1σ would be expected to encompass 68% of the data. Table C-3 indicates that six of the eight (75%) surveillance data points fall inside the +/- 1σ of 17°F scatter band for surveillance base metals. Therefore, all the plate data are deemed "credible" per the third criterion.

Table C-3 indicates that all surveillance data points fall inside the +/- 1σ of 28°F scatter band for surveillance weld materials. 100% of the data are bounded; therefore, the surveillance weld data are deemed "credible" per the third criterion. Note that, while credible, the data for surveillance weld Heat # 13253 was determined not to be relevant to the welds in the reactor vessel; therefore, the data are not used for embrittlement calculations of the Salem Unit 2 reactor vessel.

Evaluation of Weld Data from All Sources (Case 5)

Next, data from all sources are considered in order to evaluate the credibility of Heat # 21935/12008 used in the Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442 A, 3-442 B, and 3-442 C using the NRC Case 5 guidelines.

WCAP-17315-NP [Ref. C-6] already concluded that the data for the Diablo Canyon Unit 2 surveillance weld are credible, therefore, the data are not re-analyzed separately herein.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pad. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Salem Unit 2 surveillance program does not include correlation monitor material. Therefore, this criterion is not applicable to Salem Unit 2.

CONCLUSION:

Based on the preceding responses to the five criteria of Regulatory Guide 1.99, Revision 2 [Ref. C-1], Section B, the Salem Unit 2 surveillance data are credible.

APPENDIX C REFERENCES:

C-1 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988. *[ADAMS Accession Number ML003740284]*

C-2 Code of Federal Regulations 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

C-3 K. Wichman, M. Mitchell, and A. Hiser, U.S. NRC Presentation, "Generic Letter 92-01 and RPV Integrity Assessment, Status, Schedule, and Issues," NRC/Industry Workshop on RPV Integrity Issues, February 1998. *[ADAMS Accession Number ML110070570]*

C-4 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

C-5 Westinghouse Report WCAP-15692, Revision 0, "Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," August 2001.

C-6 Westinghouse Report WCAP-17315-NP, Revision 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," July 2011.

Enclosure 3

Westinghouse WCAP-16982-NP, Rev. 3, Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity (Non-Proprietary)

WCAP-16982-NP
Revision 3

December 2020

Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity

Westinghouse

WCAP-16982-NP
Revision 3

Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity

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December 2020

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RECORD OF REVISION

Revision 0:	Original Issue
Revision 1:	This revision incorporates instrument uncertainties into the Leak Test curves and updates the Pressure-Temperature (P-T) limit curves to account for vacuum refill.
Revision 2:	This revision corrects a typographical error on the x-axis of Figures 6-1 and 6-2.
Revision 3:	This revision updates the minimum criticality temperature in Figure 6-1 and Table 6-2 to be consistent with the minimum temperature for the leak test in Table 6-2 (289°F). Changes are marked with revision bars.

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EXECUTIVE SUMMARY

This report presents the Time Limited Aging Analyses for the Salem Unit 2 reactor pressure vessel in accordance with the requirements of the License Renewal Rule, 10 CFR Part 54. Time Limited Aging Analyses are calculations which evaluate some safety-related aspects of the reactor pressure vessel within the bounds of the current 40-year license that must be re-done to account for an extended period of operation.

A summary of results for the Salem Unit 2 TLAA is provided below. Based on the TLAA results, it is concluded that the Salem Unit 2 reactor vessel will remain adequate through the extended period of operation.

Identification of Extended Beltline Materials

The additional or new vessel materials that exceed 1.0E+17 n/cm^2 (E > 1.0 MeV) fluence at End-Of-License Renewal (EOLR) that were not considered in previous evaluations are the following:

- Upper Shell Plates B4711-1, B4711-2, and B4711-3
- Upper Shell Longitudinal Weld Seams 1-442 A, B, and C
- Intermediate Shell to Upper Shell Circumferential Weld Seam 8-442

These are considered to be the extended beltline materials. See Section 2 for more details.

EOLR PTS Values

All of the Salem Unit 2 reactor vessel materials that exceed a surface fluence of 1.0E+17 n/cm^2 (E > 1.0 MeV) at 50 EFPY are below the RT_{PTS} screening criteria values of 270°F, for axially oriented welds and plates\forgings, and 300°F, for circumferentially oriented welds, at 50 EFPY. See Section 4 for more details.

EOLR USE Values

All of the beltline materials in the Salem Unit 2 reactor vessel are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50 Appendix G) at 50 EFPY. See Section 5 for more details.

EOLR ART Values

The Lower Shell Longitudinal Weld Seams 3-442 A&C resulted in the highest ART value at the 1/4 vessel thickness (T) and the 3/4T locations. These limiting ART values are summarized below in Table 6-1 of Section 6. See Section 6 for more details.

Heatup and Cooldown P-T Limit Curves Applicable to EOLR

New heatup and cooldown P-T limit curves, applicable to 50 EFPY, were developed based on the limiting 1/4T and 3/4T ART values shown in Table 6-1 of Section 6. The resulting heatup and cooldown P-T limit curves are shown in Figures 6-1 and 6-2, respectively, with the corresponding data points shown in Tables 6-2 and 6-3. See Section 6 for more details. Furthermore, Appendix A provides the thermal stress intensity factors (K_{IT}) for the maximum heatup and cooldown rates at 50 EFPY in Tables A-1 and A-2, respectively.

LTOP System Enable Temperature

The minimum required LTOP system enable temperature (without margins for temperature uncertainty) for the Salem Unit 2 reactor vessel was conservatively chosen to be 274°F for 50 EFPY. See Section 6 for more details.

Surveillance Capsule Withdrawal Schedule

The maximum EOLR shift in transition temperature of the Salem Unit 2 reactor vessel materials that exceed a surface fluence of 1.0E+17 n/cm^2 (E > 1.0 MeV) is 229.3°F for 50 EFPY. Therefore, per Table 1 of ASTM E185-82, the recommended minimum number of surveillance capsules to be withdrawn is five. The new conclusion of a five capsule withdrawal schedule is based on a 60 year license. The new withdrawal schedule is shown in Table 7-1 of Section 7. See Section 7 for more details.

ERG P-T Limit Categorization

The Salem Unit 2 reactor vessel material with the highest RT_{NDT} value is the lower shell plate longitudinal weld seams 3-442 A&C. Thus, this material is considered the limiting material for Salem Unit 2. The lower shell plate longitudinal weld seams 3-442 A&C have an EOLR RT_{PTS} (RT_{NDT}) value of 239°F (based on Table 4-1 of Section 4). The EOLR RT_{PTS} (RT_{NDT}) value of 239°F places Salem Unit 2 in Category II.

The operating time at which Salem Unit 2 would transition from ERG P-T limit Category I to Category II was determined to be approximately **24.6 EFPY**. See Section 8 for more details.

1 TIME LIMITED AGING ANALYSIS

Time-limited aging analyses (TLAAs) are those licensee calculations that:

- Consider the effects of aging
- Involve time-limited assumptions defined by the current operating term, for example, 40 years
- Involve systems, structures, and components (SSCs) within the scope of license renewal
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform its intended functions
- Were determined to be relevant by the licensee in making a safety determination
- Are contained or incorporated by reference in the current licensing basis (CLB)

The potential TLAAs for the reactor pressure vessel (RPV) are identified in Table 1-1 along with indication of whether or not they meet the six criteria of 10 CFR 54.3 [Ref. 1] for TLAAs.

Table 1-1 Evaluation of Time-Limited Aging Analyses Per the Criteria of 10 CFR 54.3

Time-Limited Aging Analysis	Calculated Beltline Fluence	Pressurized Thermal Shock	Upper Shelf Energy	Pressure Limits for Heatup and Cooldown
Considers the Effects of Aging	YES	YES	YES	YES
Involves Time-Limited Assumptions Defined by the Current Operating Term	YES	YES	YES	YES
Involves SSC Within the Scope of License Renewal	YES	YES	YES	YES
Involves Conclusions or Provides the Basis for Conclusions Related to the Capability of SSC to Perform Its Intended Function	YES	YES	YES	YES
Determined to be Relevant by the Licensee in Making a Safety Determination	YES	YES	YES	YES
Contained or Incorporated by Reference in the CLB	YES	YES	YES	YES

2 CALCULATED BELTLINE FLUENCE

At currently licensed service times and operating conditions, the Salem Unit 2 RPV fracture toughness properties provide adequate margins of safety against vessel failure. However, as a vessel accumulates more and more service time, neutron irradiation (fluence) reduces material fracture toughness and initial safety margins. Prevention of RPV failure depends primarily on maintaining RPV material fracture toughness at levels that resist brittle fracture during plant operation. The first step in the TLAA of vessel embrittlement is the calculation of the neutron fluence that causes the embrittlement to increase with time.

The reactor vessel beltline neutron fluence values applicable to a postulated 20-year license renewal period were calculated for each of the Salem Unit 2 RPV beltline materials. The analysis methods used to calculate the Salem Unit 2 vessel fluences satisfy the requirements set forth in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. 2].

In accordance with Item IV.A2-23 of NUREG-1801, Volume 2, Revision 1 [Ref. 3], any materials exceeding 1.0E+17 n/cm^2 (E > 1.0 MeV) must be monitored to evaluate the changes in fracture toughness. RPV materials that are not traditionally thought of as being plant limiting because of low levels of neutron radiation must now be evaluated to determine the accumulated fluence at 50 EFPY. Therefore, fluence calculations were performed for the Salem Unit 2 RPV upper shell and nozzle forgings to determine if they will exceed 1.0E+17 n/cm^2 (E > 1.0 MeV) at 50 EFPY. The materials that exceed this threshold are referred to as extended beltline materials in this report and are evaluated to determine their impact to the proposed license renewal period.

In all cases, the maximum exposure occurs at the radial location of the pressure vessel clad/base metal interface. Data is given for the nominal end of Cycle 15 (16.1 Effective Full Power Years (EFPY)) as well as for projections through 50 EFPY of reactor operation. Projections for future operation were based on the continued use of the Cycle 15 spatial power distribution and a core power level of 3459 MWt. Table 2-1 summarizes the maximum projected neutron fluence for each of the reactor pressure vessel beltline region materials. From Table 2-1, it is noted that, although the upper shell course and associated longitudinal welds and the upper shell to intermediate shell circumferential weld are projected to exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) threshold, the nozzles themselves as well as the nozzle to upper shell welds remain below 1.0E+17 n/cm^2 (E > 1.0 MeV) through 50 EFPY of operation. Likewise, the lower shell to lower head circumferential weld remains out of the beltline region through 50 EFPY.

The fluence analysis of the upper shell and nozzles revealed that additional materials will exceed the 1.0E+17 n/cm^2 (E > 1.0 MeV) threshold. These extended beltline materials, including the respective ID's and heat numbers, are shown in Table 2-2.

Table 2-1 Salem Unit 2 Calculated Neutron Fluence Projections at the Peak Location on the Reactor Vessel Clad/Base Metal Interface [n/cm^2, (E > 1.0 MeV)]

Operating Time [efpy]	Lower Shell Plates	Low.-Int. Shell Circ. Weld	Int. Shell Plates	Int.-Upper Shell Circ. Weld	Upper Shell Plates	Lower Shell Long. Weld 60/300	Lower Shell Long. Weld 180
16.1	7.08E+18	7.08E+18	7.07E+18	1.18E+17	1.18E+17	5.06E+18	2.86E+18
17.4	7.58E+18	7.58E+18	7.57E+18	1.27E+17	1.27E+17	5.43E+18	3.04E+18
18.8	8.08E+18	8.08E+18	8.07E+18	1.36E+17	1.36E+17	5.79E+18	3.22E+18
20.0	8.52E+18	8.52E+18	8.50E+18	1.44E+17	1.44E+17	6.12E+18	3.37E+18
22.0	9.25E+18	9.25E+18	9.23E+18	1.57E+17	1.57E+17	6.65E+18	3.63E+18
24.0	9.99E+18	9.98E+18	9.96E+18	1.70E+17	1.70E+17	7.19E+18	3.89E+18
26.0	1.07E+19	1.07E+19	1.07E+19	1.83E+17	1.83E+17	7.73E+18	4.14E+18
28.0	1.15E+19	1.15E+19	1.14E+19	1.96E+17	1.96E+17	8.27E+18	4.40E+18
30.0	1.22E+19	1.22E+19	1.22E+19	2.09E+17	2.09E+17	8.81E+18	4.66E+18
32.0	1.29E+19	1.29E+19	1.29E+19	2.22E+17	2.22E+17	9.35E+18	4.92E+18
34.0	1.37E+19	1.37E+19	1.36E+19	2.36E+17	2.36E+17	9.89E+18	5.18E+18
36.0	1.44E+19	1.44E+19	1.43E+19	2.49E+17	2.49E+17	1.04E+19	5.43E+18
38.0	1.52E+19	1.51E+19	1.51E+19	2.62E+17	2.62E+17	1.10E+19	5.70E+18
40.0	1.59E+19	1.59E+19	1.58E+19	2.75E+17	2.75E+17	1.15E+19	5.96E+18
42.0	1.66E+19	1.66E+19	1.65E+19	2.88E+17	2.88E+17	1.21E+19	6.22E+18
44.0	1.74E+19	1.73E+19	1.73E+19	3.01E+17	3.01E+17	1.26E+19	6.48E+18
46.0	1.81E+19	1.81E+19	1.80E+19	3.14E+17	3.14E+17	1.32E+19	6.74E+18
48.0	1.89E+19	1.89E+19	1.87E+19	3.28E+17	3.28E+17	1.37E+19	7.01E+18
50.0	1.96E+19	1.96E+19	1.95E+19	3.41E+17	3.41E+17	1.43E+19	7.27E+18

Operating Time [efpy]	Int. Shell Long. Weld 0	Int. Shell Long. Weld 120/240	Upper Shell Long. Weld 60/300	Upper Shell Long. Weld 180	Inlet and Outlet Nozzles	Nozzle to Shell Welds	Lower Shell to Lower Head Weld
16.1	2.86E+18	5.05E+18	8.43E+16	4.78E+16	<1.0E+17	<1.0E+17	<1.0E+17
17.4	3.04E+18	5.42E+18	9.08E+16	5.09E+16	<1.0E+17	<1.0E+17	<1.0E+17
18.8	3.21E+18	5.78E+18	9.74E+16	5.41E+16	<1.0E+17	<1.0E+17	<1.0E+17
20.0	3.37E+18	6.11E+18	1.03E+17	5.69E+16	<1.0E+17	<1.0E+17	<1.0E+17
22.0	3.62E+18	6.64E+18	1.13E+17	6.15E+16	<1.0E+17	<1.0E+17	<1.0E+17
24.0	3.88E+18	7.18E+18	1.23E+17	6.61E+16	<1.0E+17	<1.0E+17	<1.0E+17
26.0	4.14E+18	7.71E+18	1.32E+17	7.07E+16	<1.0E+17	<1.0E+17	<1.0E+17
28.0	4.39E+18	8.25E+18	1.42E+17	7.53E+16	<1.0E+17	<1.0E+17	<1.0E+17
30.0	4.65E+18	8.78E+18	1.51E+17	8.00E+16	<1.0E+17	<1.0E+17	<1.0E+17
32.0	4.91E+18	9.32E+18	1.61E+17	8.46E+16	<1.0E+17	<1.0E+17	<1.0E+17
34.0	5.16E+18	9.85E+18	1.71E+17	8.92E+16	<1.0E+17	<1.0E+17	<1.0E+17
36.0	5.42E+18	1.04E+19	1.80E+17	9.38E+16	<1.0E+17	<1.0E+17	<1.0E+17
38.0	5.68E+18	1.09E+19	1.90E+17	9.85E+16	<1.0E+17	<1.0E+17	<1.0E+17
40.0	5.93E+18	1.15E+19	2.00E+17	1.03E+17	<1.0E+17	<1.0E+17	<1.0E+17
42.0	6.19E+18	1.20E+19	2.09E+17	1.08E+17	<1.0E+17	<1.0E+17	<1.0E+17
44.0	6.45E+18	1.25E+19	2.19E+17	1.12E+17	<1.0E+17	<1.0E+17	<1.0E+17
46.0	6.70E+18	1.31E+19	2.29E+17	1.17E+17	<1.0E+17	<1.0E+17	<1.0E+17
48.0	6.96E+18	1.36E+19	2.38E+17	1.22E+17	<1.0E+17	<1.0E+17	<1.0E+17
50.0	7.22E+18	1.41E+19	2.48E+17	1.26E+17	<1.0E+17	<1.0E+17	<1.0E+17

Table 2-2 Summary of Salem Unit 2 RPV Extended Beltline Materials

Salem Unit 2		
Material	**Material ID**	**Heat Number**
Upper Shell	B4711-1	C4194-1
Upper Shell	B4711-2	C4149-2
Upper Shell	B4711-3	C4171-1
Upper Shell Longitudinal Weld Seams	1-442 A, B & C	20291/1P2809 Linde 1092, Lot 3854
Intermediate to Upper Shell Circumferential Weld Seam	8-442	90099, Linde 0091, Lot 3977

3 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan [Ref. 4]. The beltline and extended beltline material properties of the Salem Unit 2 reactor vessel are presented in Table 3-1. Note that the closure head and vessel flanges were included in Table 3-1 as these materials are considered for the minimum temperature requirement of 10 CFR 50, Appendix G [Ref. 5] for pressure-temperature limits. The minimum temperature pertains to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature.

The chemistry factors (CFs) were calculated using Positions 1.1 and 2.1 of Regulatory Guide 1.99, Rev. 2 [Ref. 6]. Position 1.1 uses Tables 1 and 2 from the Reg. Guide along with the best estimate copper and nickel weight percents, which are presented in Table 3-1 of this report. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The calculated fluence values at the surveillance capsule locations are provided in Table 3-2 and are used to determine the CFs in Table 3-3. The capsule fluence values were determined using ENDF/B-VI cross-sections and followed the guidance in Regulatory Guide 1.190 [Ref. 2]. Table 3-4 summarizes the Positions 1.1 and 2.1 CFs determined for the Salem Unit 2 RPV beltline and extended beltline materials.

Table 3-1 Best Estimate Cu and Ni Weight Percent, Initial RT_{NDT} Values and Initial USE Values for the Salem Unit 2 RPV Beltline and Extended Beltline Materials

RPV Material Description	Material Heat Number	Cu (%)	Ni (%)	Initial USE (ft-lb)	Initial RT_{NDT} (°F)
Closure Head Flange B4702-1[(a)]	---	---	---	---	-40
Vessel Flange B5001[(a)]	---	---	---	---	12
Upper Shell B4711-1[(b)]	C4194-1	0.11	0.55	87.1[(c)]	60
Upper Shell B4711-2[(b)]	C4149-2	0.14	0.56	79.3[(d)]	60
Upper Shell B4711-3[(b)]	C4171-1	0.12	0.58	69.3[(e)]	101
Intermediate Shell B4712-1[(f)]	C4173-1	0.13	0.56	106	0
Intermediate Shell B4712-2[(f)]	C4186-2	0.12	0.61	97	12
Intermediate Shell B4712-3[(f)]	C4194-2	0.11	0.57	107	10
Lower Shell B4713-1[(f)]	C4182-1	0.12	0.60	98	8
Lower Shell B4713-2[(f)]	C4182-2	0.12	0.57	103	8
Lower Shell B4713-3[(f)]	B-8343-1	0.12	0.58	121	10
Intermediate to Upper Shell Circumferential Weld Seam 8-442	20291/1P2809 Linde 1092, Lot 3854	0.27[(g)]	0.735[(g)]	97[(h)]	-56[(i)]
Intermediate to Lower Shell Circumferential Weld Seam 9-442[(f)]	90099, Linde 0091, Lot 3977	0.197	0.060	99.7	-56[(i)]
Upper Shell Longitudinal Weld Seams 1-442 A, B & C	13253/12008, Linde 1092, Lot 3833	0.21[(g)]	0.873[(g)]	97[(h)]	-56[(i)]
Intermediate Shell Longitudinal Weld Seams 2-442 A, B & C[(f)]	13253/20291, Linde 1092, Lot 3833	0.221	0.732	96.2	-56[(i)]
Lower Shell Longitudinal Weld Seams 3-442 A, B & C[(f)]	21935/12008, Linde 1092, Lot 3889	0.213	0.867	114	-56[(i)]
Surveillance Weld Material[(f)]	13253, Linde 1092 Lot 3833/3774	0.225	0.727	---	---

Notes:

(a) The initial RT_{NDT} value for the vessel flange was obtained from WCAP-15693 [Ref. 7]. Due to the reactor vessel head replacement, the closure head flange was also replaced and the new initial RT_{NDT} value was provided in S-TODI-2008-0010 [Ref. 8].

(b) Values from CMTR-RV-PNJ [Ref. 9]. Note that the initial RT_{NDT} values were determined in accordance with the requirements of Subparagraph NB-2331 of Section III of the ASME B&PV Code [Ref. 10], as specified by Paragraph II - D of 10 CFR Part 50, Appendix G [Ref. 5]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB Section II.5-2 ("Fracture Toughness") of the NRC Regulatory Standard Review Plan [Ref. 4]. Since transversely oriented Charpy V-Notch specimens were not tested for the upper shell plates, the initial RT_{NDT} was determined using Section 1.1(3)(a) and (b).

(c) Average USE value calculated based on ASTM E185-82 [Ref. 11] and Charpy test data points documented in CMTR-RV-PNJ [Ref. 9]: 133 and 135 ft-lbs. Due to specimen orientation, this average value has been conservatively reduced by 65%.

(d) Average USE value calculated based on ASTM E185-82 [Ref. 11] and Charpy test data points documented in CMTR-RV-PNJ [Ref. 9]: 123,125, and 118 ft-lbs. Due to specimen orientation, this average value has been conservatively reduced by 65%.

(e) Average USE value calculated based on ASTM E185-82 [Ref. 11] and Charpy test data points documented in CMTR-RV-PNJ [Ref. 9]: 106,105, and 109 ft-lbs. Due to specimen orientation, this average value has been conservatively reduced by 65%.

(f) Values from WCAP-15693 [Ref. 7].

(g) Best estimate copper and nickel content based on CE NPSD-1119, Revision 1 [Ref. 12]. Note that in the case of Intermediate to Upper Shell Circumferential Weld Seam 8-442, there is no record that the two heats of 20291 and 1P2809 were deposited in tandem; therefore, the copper and nickel content for heat 1P2809 is used as a more conservative basis for estimating the effect of neutron irradiation.

(h) Generic upper shelf energy values are based on CEN-622-A [Ref. 13].

(i) Generic initial RT_{NDT} values per 10 CFR 50.61 [Ref. 14].

Table 3-2 Calculated Integrated Neutron Exposure of the Surveillance Capsules at Salem Unit 2

Capsule	Fluence
T	2.75E+18 n/cm^2, (E > 1.0 MeV)
U	5.82E+18 n/cm^2, (E > 1.0 MeV)
X	1.12E+19 n/cm^2, (E > 1.0 MeV)
Y	1.82E+19 n/cm^2, (E > 1.0 MeV)

Table 3-3 Calculation of CF Values using Salem Unit 2 Surveillance Capsule Test Results

RPV Material	Capsule	Capsule Fluence[(a)]	FF[(b)]	ΔRT_{NDT}[(c)]	FF*ΔRT_{NDT}	FF^2
		(E+19 n/cm^2)		°F	°F	
Intermediate Shell B4712-2 (Longitudinal) [(c)]	T	0.275	0.648	61.66	39.96	0.420
	U	0.582	0.849	66.54	56.46	0.720
	X	1.12	1.032	93.82	96.79	1.064
	Y	1.82	1.164	105.69	123.05	1.356
Intermediate Shell B4712-2 (Transverse) [(c)]	T	0.275	0.648	74.83	48.49	0.420
	U	0.582	0.849	98.26	83.37	0.720
	X	1.12	1.032	125.15	129.11	1.064
	Y	1.82	1.164	129.33	150.58	1.356
				Sum =	727.81	7.119
	CF = ∑(FF * ΔRT_{NDT}) / ∑(FF^2) = (727.81 / 7.119) = 102.2°F					
Surveillance Weld Material[(d)]	T	0.275	0.648	153.17	99.25	0.420
	U	0.582	0.849	185.94	157.77	0.720
	X	1.12	1.032	195.43	201.62	1.064
	Y	1.82	1.164	200.90	233.90	1.356
				Sum =	692.55	3.560
	CF = ∑(FF * ΔRT_{NDT}) / ∑(FF^2) = (692.55 / 3.560) = 194.5°F					

Notes:

(a) f = calculated fluence values from Table 3-2.

(b) FF = fluence factor = $f^{(0.28-0.1*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values from Table 4-6 of WCAP-15693 [Ref. 7].

(d) Since the surveillance weld metal is only representative of the beltline welds, and not identical, it is not used in any of the calculations documented in this report. Thus, there is no Position 2.1 CF value available for the beltline welds. This is consistent with WCAP-15693 [Ref. 7].

Table 3-4 Summary of the Salem Unit 2 RPV Beltline and Extended Beltline Material Chemistry Factors based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

RPV Material	Chemistry Factor (°F)	
	Position 1.1	Position 2.1
Upper Shell B4711-1	73.5	---
Upper Shell B4711-2	98.2	---
Upper Shell B4711-3	82.6	---
Intermediate Shell B4712-1	89.8	---
Intermediate Shell B4712-2	83.2	102.2
Intermediate Shell B4712-3	73.7	---
Lower Shell B4713-1	83.0	---
Lower Shell B4713-2	82.4	---
Lower Shell B4713-3	82.6	---
Intermediate to Upper Shell Circumferential Weld Seam 8-442 (Heat # 1P2809)	205.6	---
Intermediate to Lower Shell Circumferential Weld Seam 9-442 (Heat # 90099)	91.4	---
Upper Shell Longitudinal Weld Seams 1-442 A, B & C (Heat # 13253/12008)	208.7	---
Intermediate Shell Longitudinal Weld Seams 2-442 A, B & C (Heat # 13253/20291)	189.1	---
Lower Shell Longitudinal Weld Seams 3-442 A, B & C (Heat # 21935/12008)	208.6	---

4 PRESSURIZED THERMAL SHOCK

A limiting condition on RPV integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the RPV under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high repressurization; significant degradation of vessel material toughness caused by radiation embrittlement; and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling (10 CFR 50.61) on PTS [Ref. 14] that established screening criteria on PWR vessel embrittlement, as measured by the maximum nil ductility reference temperature in the limiting beltline component, termed RT_{PTS}. RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 [Ref. 6].

These accepted methods were used with the surface fluence of Section 2 to calculate the following RT_{PTS} values for the Salem Unit 2 RPV materials at 50 EFPY, which is the end of the license renewal period (see Table 4-1).

PTS Conclusion

The limiting RT_{PTS} value for the axially oriented welds and plates is 239°F (see Table 4-1); this value corresponds to the Lower Shell Longitudinal Weld Seams 3-442 A&C.

The limiting RT_{PTS} value for the circumferentially oriented welds is 118°F (see Table 4-1); this value corresponds to the Intermediate Shell to Lower Shell Circumferential Weld Seam 9-442.

Therefore, all of the Salem Unit 2 reactor vessel materials that exceed a surface fluence of 1.0E+17 n/cm^2 (E > 1.0 MeV) at 50 EFPY are below the RT_{PTS} screening criteria values of 270°F, for axially oriented welds and plates/forgings, and 300°F, for circumferentially oriented welds, at 50 EFPY.

Table 4-1 Calculation of RT_{PTS} Values for 50 EFPY at the Clad/Base Metal Interface

RPV Material	CF[a] (°F)	Fluence[b] (E+19 n/cm^2)	FF[c]	ΔRT_{PTS}[d] (°F)	$RT_{NDT(U)}$[e] (°F)	σ_U (°F)	σ_Δ [g] (°F)	M (°F)	RT_{PTS} (°F)
Upper Shell B4711-1	73.5	0.0341	0.237	17.39	60	0	8.69	17.39	95
Upper Shell B4711-2	98.2	0.0341	0.237	23.23	60	0	11.61	23.23	106
Upper Shell B4711-3	82.6	0.0341	0.237	19.54	101	0	9.77	19.54	140
Intermediate Shell B4712-1	89.8	1.95	1.182	106.19	0	0	17	34	140
Intermediate Shell B4712-2 (without credible surveillance data)	*83.2*	*1.95*	*1.182*	*98.38*	*12*	*0*	*17*	*34*	*144*
Intermediate Shell B4712-2 (with credible surveillance data)	102.2	1.95	1.182	120.85	12	0	8.5	17	150
Intermediate Shell B4712-3	73.7	1.95	1.182	87.15	10	0	17	34	131
Lower Shell B4713-1	83.0	1.96	1.184	98.26	8	0	17	34	140
Lower Shell B4713-2	82.4	1.96	1.184	97.55	8	0	17	34	140
Lower Shell B4713-3	82.6	1.96	1.184	97.78	10	0	17	34	142
Intermediate to Upper Shell Circumferential Weld Seam 8-442 (Heat # 20291/1P2809)	205.6	0.0341	0.237	48.63	-56[f]	17	24.32	59.34	52
Intermediate to Lower Shell Circumferential Weld Seam 9-442 (Heat # 90099)	91.4	1.96	1.184	108.20	-56[f]	17	28	65.51	118
Upper Shell Longitudinal Weld Seams 1-442 A&C (Heat # 13253/12008)	208.7	0.0248	0.196	40.94	-56[f]	17	20.47	53.22	38
Upper Shell Longitudinal Weld Seam 1-442 B (Heat # 13253/12008)	208.7	0.0126	0.128	26.72	-56[f]	17	13.36	43.24	14

Table 4-1 Calculation of RT_{PTS} Values for 50 EFPY at the Clad/Base Metal Interface

RPV Material	CF[(a)] (°F)	Fluence[(b)] (E+19 n/cm^2)	FF[(c)]	ΔRT_{PTS}[(d)] (°F)	$RT_{NDT(U)}$[(e)] (°F)	σ_U (°F)	σ_Δ[(g)] (°F)	M (°F)	RT_{PTS} (°F)
Intermediate Shell Longitudinal Weld Seam 2-442 A (Heat # 13253/20291)	189.1	0.722	0.909	171.82	-56[(f)]	17	28	65.51	181
Intermediate Shell Longitudinal Weld Seam 2-442 B&C (Heat # 13253/20291)	189.1	1.41	1.095	207.13	-56[(f)]	17	28	65.51	217
Lower Shell Longitudinal Weld Seams 3-442 A&C (Heat # 21935/12008)	208.6	1.43	1.099	229.30	-56[(f)]	17	28	65.51	**239**
Lower Shell Longitudinal Weld Seam 3-442 B (Heat # 21935/12008)	208.6	0.727	0.911	189.94	-56[(f)]	17	28	65.51	199

Notes:

(a) From Table 3-4 of this report.
(b) From Table 2-1 of this report.
(c) FF = fluence factor = $f^{(0.28-0.1*\log(f))}$.
(d) ΔRT_{PTS} = CF * FF.
(e) From Table 3-1 of this report. All $RT_{NDT(U)}$ values are measured values unless otherwise noted. Note that σ_U = 0°F for measured values.
(f) $RT_{NDT(U)}$ values are generic mean values per 10 CFR 50.61. Note that σ_U = 17°F for generic values.
(g) Per WCAP-15692, Rev. 0 [Ref. 15], the surveillance plate and weld data obtained for Salem Unit 2 has been deemed credible. Per the guidance of Reg. Guide 1.99, Revision 2 [Ref. 6], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8.5°F for Position 2.1; and the weld metal σ_Δ = 28°F for Position 1.1 and σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed 0.5*ΔRT_{NDT}. Note that the Position 2.1 is not used for the surveillance weld data as the surveillance weld is not identical to the beltline welds (this is consistent with WCAP-15693 [Ref. 7]).

5 UPPER SHELF ENERGY

The decrease in upper shelf Charpy energy is associated with the determination of acceptable RPV toughness during the license renewal period when the vessel is exposed to additional irradiation.

The requirements on upper shelf energy are included in 10 CFR 50, Appendix G [Ref. 5]. 10 CFR 50, Appendix G requires utilities to submit an analysis at least 3 years prior to the time that the upper shelf energy of any of the RPV material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

There are two methods that can be used to estimate the change in upper shelf energy (USE) with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99, Revision 2. For vessel beltline materials that are not in the surveillance program or not credible, the Charpy USE is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2 [Ref. 6].

When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the Regulatory Guide data to predict the change in USE of the RPV due to irradiation.

Using the 1/4T fluence values, projected upper shelf energy values were calculated to determine if the Salem Unit 2 beltline and extended beltline materials remain above the 50 ft-lb limit at 50 EFPY (see Tables 5-1 and 5-2).

USE Conclusion

All of the beltline and extended beltline materials in the Salem Unit 2 reactor vessel are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50 Appendix G) at 50 EFPY.

Table 5-1 Predicted Position 1.2 USE Values at 50 EFPY (EOLR)

RPV Material	Cu[(a)] (%)	1/4T Fluence[(b)] (E+19 n/cm^2)	Initial USE[(a)] (ft-lb)	USE Decrease (%)	USE (ft-lb)
Upper Shell B4711-1	0.11	0.020	87.1	9	79
Upper Shell B4711-2	0.14	0.020	79.3	10	71
Upper Shell B4711-3	0.12	0.020	69.3	9	63
Intermediate Shell B4712-1	0.13	1.162	106	25	80
Intermediate Shell B4712-2	0.12	1.162	97	25	73
Intermediate Shell B4712-3	0.11	1.162	107	25	80
Lower Shell B4713-1	0.12	1.168	98	25	74
Lower Shell B4713-2	0.12	1.168	103	25	77
Lower Shell B4713-3	0.12	1.168	121	25	91
Intermediate to Upper Shell Circumferential Weld Seam 8-442 (Heat # 1P2809)	0.27	0.020	97	18	80
Intermediate to Lower Shell Circumferential Weld Seam 9-442 (Heat # 90099)	0.197	1.168	99.7	35	65
Upper Shell Longitudinal Weld Seams 1-442 A&C (Heat # 13253/12008)	0.21	0.015	97	16[(c)]	81
Upper Shell Longitudinal Weld Seam 1-442 B (Heat # 13253/12008)	0.21	0.008	97	16[(c)]	81
Intermediate Shell Longitudinal Weld Seam 2-442 A (Heat # 13253/20291)	0.221	0.430	96.2	33	64
Intermediate Shell Longitudinal Weld Seam 2-442 B&C (Heat # 13253/20291)	0.221	0.840	96.2	38	60
Lower Shell Longitudinal Weld Seams 3-442 A&C (Heat # 21935/12008)	0.213	0.852	114	38	71
Lower Shell Longitudinal Weld Seam 3-442 B (Heat # 21935/12008)	0.213	0.433	114	33	76

Notes:

(a) From Table 3-1 of this report.

(b) 1/4 T EOLR Fluence = $f_{surf} * \exp^{(-0.24*X)}$, where x is the depth (in inches) into the vessel wall from the inner surface; for the 1/4T location, x = 0.25 * 8.625 inches = 2.156 inches.

(c) The fluence ranges from 2.0E+17 n/cm^2 to 6.0E+19 n/cm^2 in Regulatory Guide 1.99, Rev. 2, Figure 2; thus, the upper shelf energy decrease was conservatively estimated based on 2.0E+17n/cm^2.

The predicted Position 2.2 EOLR USE value is calculated in Table 5-2 for the surveillance material (i.e., Intermediate Shell Plate B4712-2). The reduced plant surveillance data from Table 5-10 of WCAP-15692 [Ref. 15] was plotted on Reg. Guide 1.99, Revision 2, Figure 2 (see Figure 5-1). This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. This reduced line was used instead of the existing lines to determine the Position 2.2 EOLR USE values.

Table 5-2 Predicted Position 2.2 USE Values at 50 EFPY (EOLR)

RPV Material	Cu (%)	1/4T Fluence (E+19 n/cm^2)	Unirradiated USE (ft-lb)	USE Decrease (%)	USE (ft-lb)
Intermediate Shell B4712-2	0.12	1.162	97	16	81

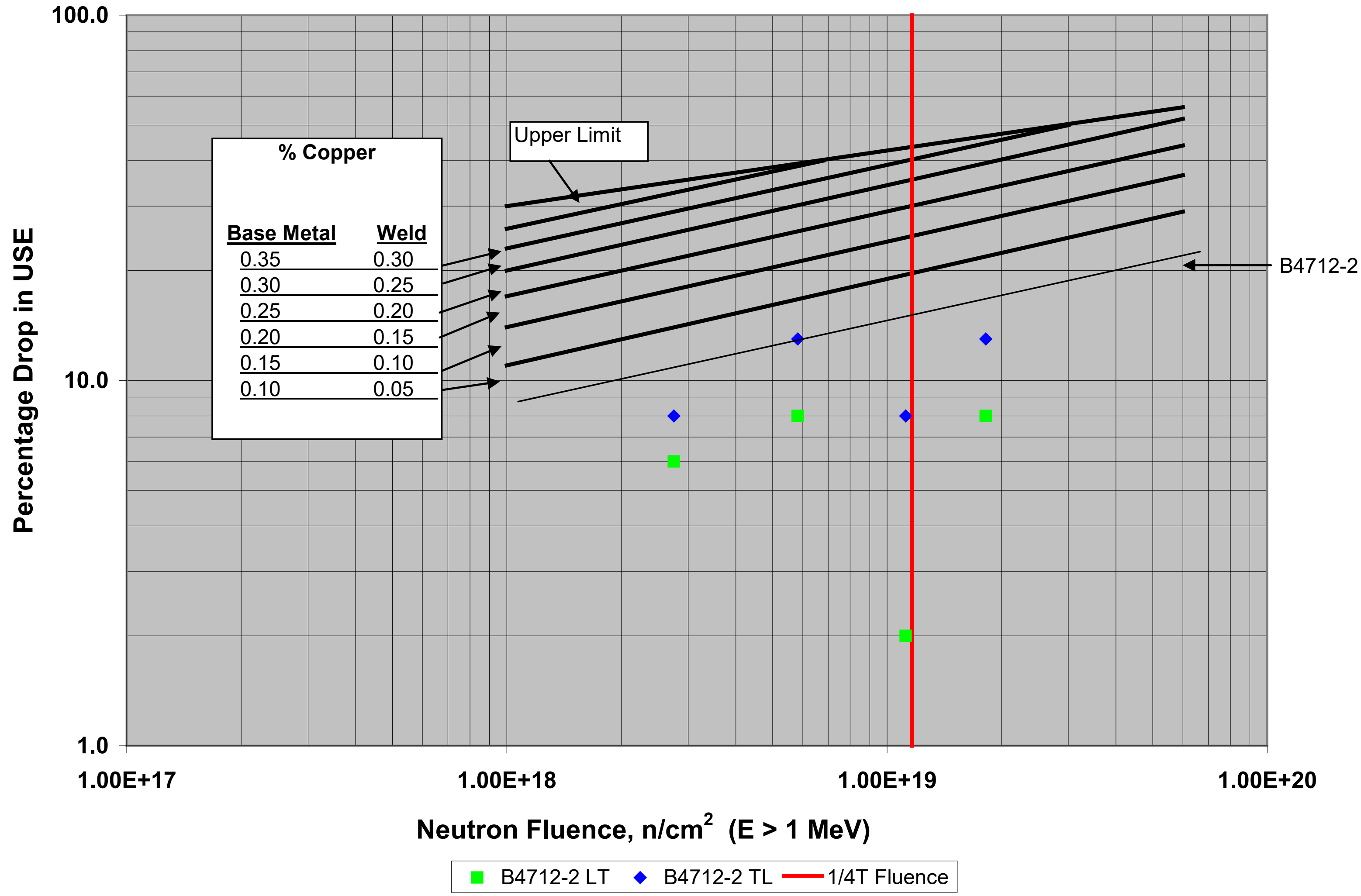

Figure 5-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence

6 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

General Design Criterion 14 of Appendix A of 10 CFR Part 50 [Ref. 16], "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage or rapid failure and of gross rupture. Likewise, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that when stressed under operating, maintenance, and testing, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. To assess the structural integrity of the RPV, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires an appropriate materials surveillance program for the RPV beltline region.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil ductility transition temperature) corresponding to the limiting material in the beltline region of the RPV. The most limiting RT_{NDT} of the material in the core (beltline) region of the RPV is determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}).

RT_{NDT} increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. Using the adjusted reference temperature (ART) values, pressure-temperature limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G [Ref. 5], as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code [Ref. 17].

The 1/4 and 3/4 thickness (1/4T and 3/4T) fluences and material properties were used to determine the limiting material and calculate its pressure-temperature limits at 50 EFPY, the end of the license renewal period being evaluated. The limiting materials were determined from the values of ART at the 1/4T and 3/4T locations and summarized in Table 6-1; Lower Shell Longitudinal Weld Seams 3-442 A and C resulted in the highest ART value at the 1/4T and 3/4T locations. These limiting ART values were calculated for an end of license renewal time of 50 EFPY.

Table 6-1 Summary of the Limiting ART Values used in Generation of the Salem Unit 2 Reactor Vessel Heatup and Cooldown Curves

EFPY	1/4 T Limiting ART	3/4 T Limiting ART
	Lower Shell Longitudinal Weld Seams 3-442 A&C	
50	209°F	150°F

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been developed utilizing the 1998 through the 2000 Summer Addenda Edition of the ASME Code Section XI, Appendix G methodology along with ASME Code Case N-641

[Ref. 17]. The 1998 through the 2000 Summer Addenda Edition of the ASME Code Section XI, Appendix G methodology allows use of the less restrictive K_{IC} stress intensity factors and also allows the use of the less restrictive "Circ-Flaw" methodology (formerly known as ASME Code Cases N-640 and N-588, respectively). Code Case N-641 provides alternative procedures for calculating the allowable pressure-temperature relationships and Low Temperature Overpressure Protection (LTOP) effective temperatures. Code Case N-641 broke Section 2215 of the 1998 through the 2000 Summer Addenda Edition of Section XI, Appendix G down into Sections 2215.1 and 2215.2 for the allowable pressures and LTOP System, respectively. Section 2215.1 of Code Case N-641 replaced all K_{IA} designations with K_{IC}, thus removing the option to use the more restrictive K_{IA} stress intensity factor. Section 2215.2 provided the methodology to determine the LTOP System effective temperature. These methodologies are documented in WCAP-14040-A, Revision 4 [Ref. 18].

Figure 6-1 presents the limiting heatup curves with margins for possible instrumentation errors [61 psi and 18°F] using heatup rates of 60 and 100°F/hr. applicable for 50 EFPY with the "Flange-Notch" requirement. Figure 6-2 presents the limiting cooldown curves with margins for possible instrumentation errors [61 psi and 18°F] using cooldown rates of 0, 20, 40, 60 and 100°F/hr. applicable for 50 EFPY with the "Flange-Notch" requirement. The corresponding data points for Figures 6-1 and 6-2 are provided in Tables 6-2 and 6-3.

6.1 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [Ref. 5] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3106 psig), which is 621 psig for Salem Unit 2. The limiting unirradiated RT_{NDT} of 12°F (due to limitations in the OPERLIM code, this temperature was conservatively rounded to the next multiple of five, 15°F) occurs in the vessel flange of the Salem Unit 2 reactor vessel, so the minimum allowable temperature of this region is 153°F at pressures greater than 560 psig (with instrument uncertainties). These limits are shown in Figure 6-1 and 6-2.

6.2 MINIMUM BOLTUP TEMPERATURE

According to WCAP-14040-A, Rev. 4 [Ref. 18], the minimum boltup temperature should be 60°F or the material RT_{NDT} of the stressed region, whichever is higher. Based on the initial RT_{NDT} values for the closure head and vessel flange in Table 3-1, the minimum boltup temperature for Salem Unit 2 is 60°F. Per S-TODI-2008-0010 [Ref. 8], a margin of 2°F was applied to the minimum boltup temperature. Thus, the minimum boltup temperature with margin is 62°F.

6.3 LTOP SYSTEM ENABLE TEMPERATURE

The minimum LTOP System enable temperature was determined to be 274°F, utilizing the methodology of ASME Code Case N-641.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL AND ART VALUES AT 50 EFPY:
1/4T ART: 209°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)
3/4T ART: 150°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)

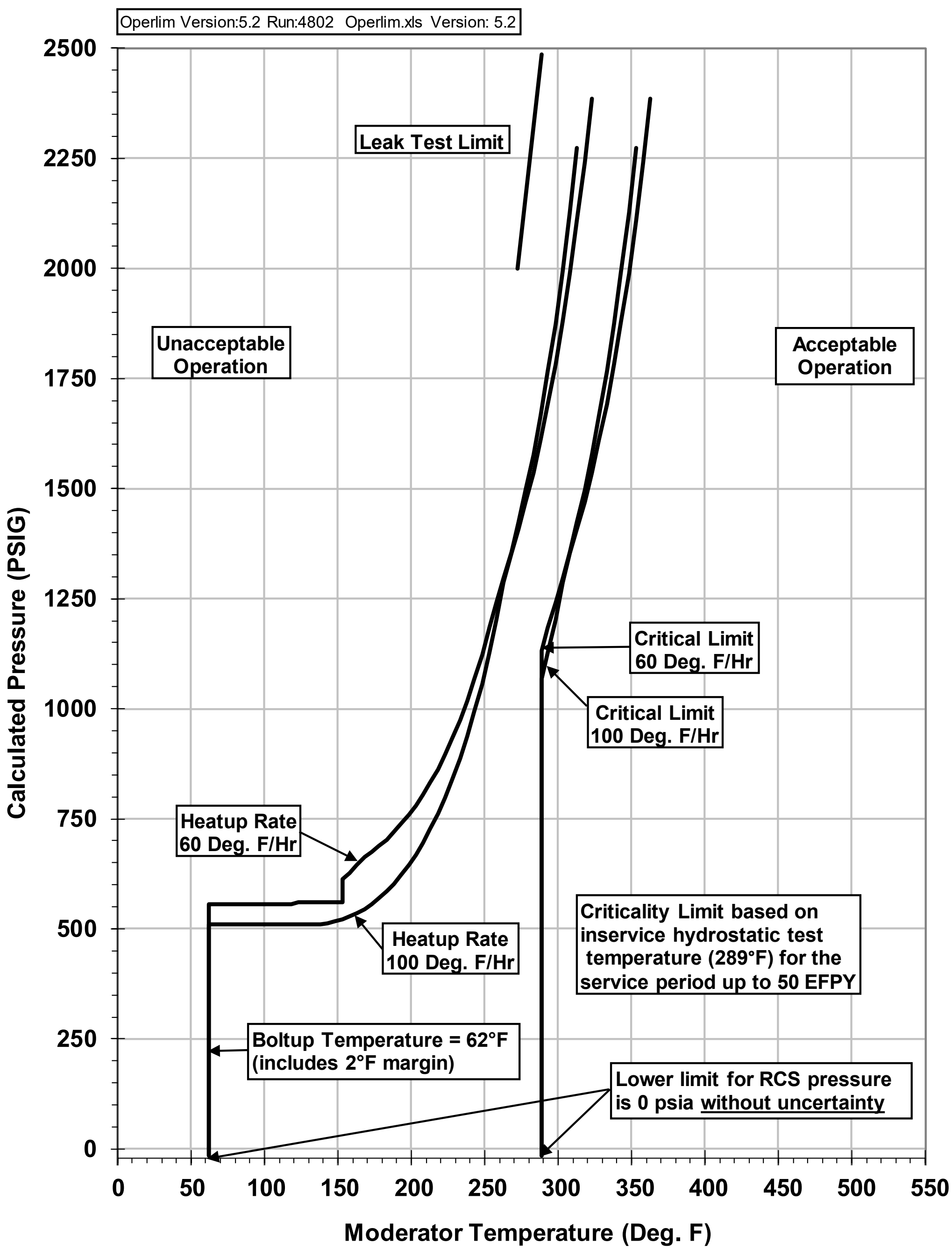

Figure 6-1 Salem Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60°F/hr. and 100°F/hr.) Applicable for 50 EFPY (with the "Flange-Notch" & with Margins for Instrumentation Errors) Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology (w/K_{IC})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL AND ART VALUES AT 50 EFPY:
1/4T ART: 209°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)
3/4T ART: 150°F (Lower Shell Longitudinal Weld Seams 3-442 A&C)

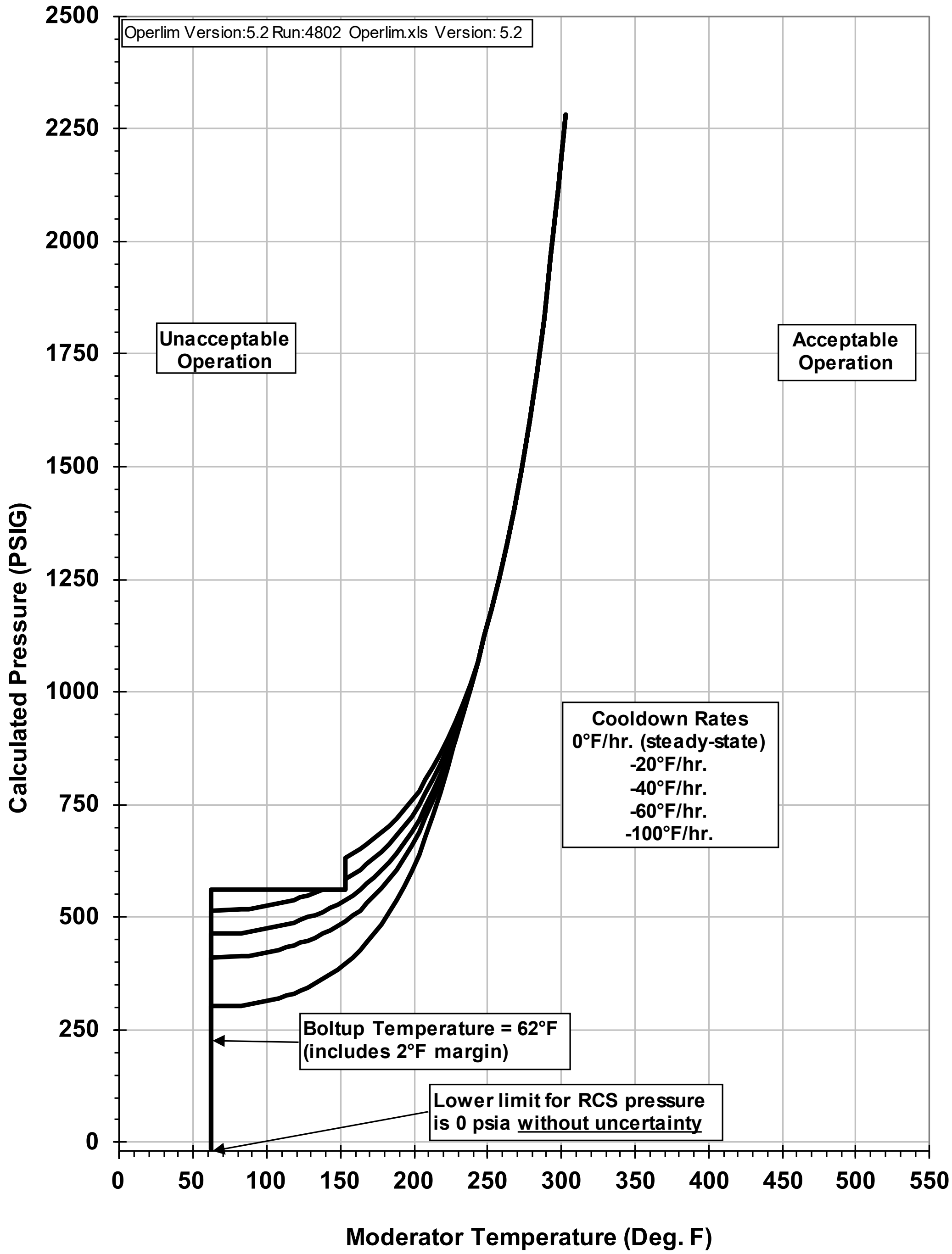

Figure 6-2 Salem Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr.) Applicable for 50 EFPY (with the "Flange-Notch" & with Margins for Instrumentation Errors) Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology (w/K_{IC})

Table 6-2 50 EFPY Heatup Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

Leak Test Limit		60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
272	2000	62	Note (a)	289	Note (a)	62	Note (a)	289	Note (a)
272	2000	62	555	289	555	62	510	289	510
289	2485	83	555	289	555	83	510	289	511
289	2485	88	555	289	556	88	510	289	511
		93	555	289	558	93	510	289	513
		98	555	289	560	98	510	289	514
		103	555	289	560	103	510	289	516
		108	555	289	560	108	510	289	518
		113	555	289	560	113	510	289	521
		118	556	289	560	118	510	289	523
		123	560	289	560	123	510	289	528
		128	560	289	560	128	510	289	531
		133	560	289	560	133	510	289	536
		138	560	289	560	138	511	289	541
		143	560	289	560	143	513	289	546
		148	560	289	560	148	516	289	552
		153	560	289	613	153	521	289	557
		153	560	289	627	153	521	289	560
		153	613	289	644	153	521	289	560
		158	627	289	663	158	528	289	560
		163	644	289	684	163	536	289	560
		168	662	289	707	168	546	289	570
		173	675	289	733	173	557	289	585
		178	688	289	762	178	570	289	603
		183	703	289	794	183	585	289	622
		188	720	289	829	188	603	289	644
		193	738	289	868	193	622	289	669
		198	758	289	906	198	644	289	696
		203	781	289	927	203	669	289	727
		208	805	289	950	208	696	289	761
		213	833	289	976	213	727	289	798
		218	863	289	1003	218	761	289	840
		223	896	289	1034	223	798	289	886
		228	933	289	1068	228	840	289	938
		233	973	289	1105	233	886	289	994
		238	1018	289	1132	238	938	289	1067
		243	1068	293	1184	243	994	293	1126
		248	1123	298	1241	248	1057	298	1203

Table 6-2 (continued) 50 EFPY Heatup Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

Leak Test Limit		60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
		253	1184	303	1296	253	1126	303	1287
		258	1241	308	1356	258	1203	308	1354
		263	1296	313	1423	263	1287	313	1409
		268	1356	318	1496	268	1354	318	1470
		273	1423	323	1577	273	1409	323	1537
		278	1496	328	1667	278	1470	328	1611
		283	1577	333	1766	283	1537	333	1692
		288	1667	338	1875	288	1611	338	1782
		293	1766	343	1995	293	1692	343	1880
		298	1875	348	2128	298	1782	348	1989
		303	1995	353	2274	303	1880	353	2109
		308	2128			308	1989	358	2241
		313	2274			313	2109	363	2386
						318	2241		
						323	2386		

Note:

(a) The lower limit for reactor coolant system pressure is 0 psia without uncertainty.

Table 6-3 50 EFPY Cooldown Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)
62	Note (a)	62	Note (a)	62	Note (a)	62	Note (a)	62	Note (a)
62	560	62	515	62	464	62	411	62	302
83	560	83	517	83	466	83	413	83	304
88	560	88	519	88	468	88	415	88	306
93	560	93	522	93	470	93	418	93	309
98	560	98	525	98	473	98	421	98	313
103	560	103	528	103	476	103	424	103	316
108	560	108	531	108	480	108	428	108	321
113	560	113	535	113	484	113	432	113	326
118	560	118	539	118	489	118	437	118	331
123	560	123	544	123	494	123	442	123	338
128	560	128	549	128	499	128	449	128	345
133	560	133	555	133	506	133	455	133	353
138	560	138	560	138	513	138	463	138	362
143	560	143	560	143	520	143	471	143	372
148	560	148	560	148	529	148	481	148	383
153	560	153	560	153	539	153	491	153	396
153	560	153	560	158	549	158	503	158	410
153	632	153	586	163	561	163	516	163	426
158	641	158	595	168	574	168	530	168	444
163	651	163	606	173	589	173	546	173	463
168	662	168	618	178	605	178	564	178	485
173	675	173	632	183	623	183	584	183	510
178	688	178	646	188	643	188	606	188	537
183	703	183	663	193	665	193	630	193	567
188	720	188	681	198	690	198	658	198	601
193	738	193	701	203	717	203	688	203	638
198	758	198	723	208	747	208	721	208	680
203	781	203	748	213	780	213	758	213	726
208	805	208	775	218	817	218	799	218	777
213	833	213	805	223	858	223	844	223	834
218	863	218	838	228	903	228	895	228	895
223	896	223	875	233	953	233	950	233	950
228	933	228	916	238	1008	238	1008	238	1008
233	973	233	961	243	1066	243	1066	243	1066
238	1018	238	1011	248	1123	248	1123	248	1123
243	1068	243	1066	253	1184	253	1184	253	1184
248	1123	248	1123	258	1251	258	1251	258	1251
253	1184	253	1184	263	1325	263	1325	263	1325
258	1251	258	1251	268	1407	268	1407	268	1407

Table 6-3 (continued) 50 EFPY Cooldown Curve Data Points Using the 1998 through 2000 Summer Addenda Edition of the ASME Boiler and Pressure Vessel Code, Section XI, App. G Methodology [With K_{IC}, With Flange, With Temperature (18°F) and Pressure (61 psi) Uncertainties, and With 2°F Margin on the Boltup Temperature]

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)	T(°F)	P (psig)
263	1325	263	1325	273	1497	273	1497	273	1497
268	1407	268	1407	278	1597	278	1597	278	1597
273	1497	273	1497	283	1708	283	1708	283	1708
278	1597	278	1597	288	1830	288	1830	288	1830
283	1708	283	1708	293	1965	293	1965	293	1965
288	1830	288	1830	298	2115	298	2115	298	2115
293	1965	293	1965	303	2280	303	2280	303	2280
298	2115	298	2115						
303	2280	303	2280						

Note:

(a) The lower limit for reactor coolant system pressure is 0 psia without uncertainty.

7 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 [Ref. 11] and is recommended for future capsules to be removed from the Salem Unit 2 reactor vessel. This recommended removal schedule is applicable to 50 EFPY of operation.

Table 7-1 Recommended Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location[(a)]	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence (n/cm^2, E > 1.0 MeV)[(c)]
T	40°	3.41	1.19	2.75E+18
U	140°	3.45	2.70	5.82E+18
X	220°	3.48	6.19	1.12E+19
Y	320°	3.47	10.80	1.82E+19
S	4°	1.38	(d)	In reactor
V	176°	1.38	(d)	In reactor
W	184°	1.38	(d)	In reactor
Z	356°	1.38	(d)	In reactor

Notes:

(a) Based on WCAP-15692 [Ref. 15].

(b) EFPY from plant startup.

(c) From Table 3-2.

(d) Capsule S, V, W, or Z could be designated as the fifth capsule to be withdrawn from the Salem Unit 2 reactor vessel to fulfill the requirements of ASTM E185-82, while the remaining three will then be standby capsules. The fifth capsule can be removed at any time after 35.4 EFPY but before the reactor vessel reaches 50 EFPY. It is recommended to be removed at 40 EFPY. At this time, the fifth capsule would have between one and two times the peak EOLR (50 EFPY) vessel fluence of 1.96E+19 n/cm^2 (E > 1.0 MeV). It is recommended that the remaining three standby capsules also be removed and placed in storage at the time the fifth capsule is withdrawn as Section X1.M31 of NUREG-1801 [Ref. 3], "Reactor Vessel Surveillance," states that any surveillance capsules that are left in the reactor vessel should provide meaningful metallurgical data. The NRC specifically states that anything beyond 60 years of exposure is not meaningful metallurgical data.

8 ERG P-T LIMITS CATEGORIZATION

The ERG limits were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event [Ref. 19]. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest value of RT_{NDT} for which the generic category ERG limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the RT_{NDT} value, which is equivalent to the RT_{PTS} value as defined in Section (a)(7) of 10 CFR 50.61 [Ref. 14]. The material with the highest RT_{NDT} defines the limiting material. The ERG limits are identified in Table 8-1.

Table 8-1 ERG P-T Limits Categories

Applicable RT_{NDT} Value[(a)]	ERG P-T Limit Category
$RT_{NDT} \leq 200°F$	Category I
$200°F < RT_{NDT} \leq 250°F$	Category II
$250°F < RT_{NDT} \leq 300°F$	Category IIIb

Note:
(a) Longitudinally oriented flaws are applicable only up to 250°F, the circumferentially oriented flaws are applicable up to 300°F.

Per the ERG limit guidance document [Ref. 19], some vessels do not change categories for operation through the end of license. However, when a vessel does change ERG categories between the beginning and end of operation, a plant-specific assessment must be performed to determine at what operating time the category changes. Thus, the ERG classification need not be changed until the operating cycle during which the maximum vessel value of actual or estimated real-time RT_{NDT} exceeds the limit on its current ERG category. *Note that for license extension, EOL in these discussions would be equal to EOLR.*

The material with the highest RT_{NDT} define the limiting material, which for Salem Unit 2 is the Lower Shell Longitudinal Weld Seams 3-442 A&C, with an EOLR RT_{PTS} (RT_{NDT}) value of 239°F (based on Table 4-1 of Section 4).

The operating time at which the ERG category would transition from Category I to Category II (i.e., when the RT_{NDT} of the lower shell longitudinal weld seams 3-442 A&C would equal 200°F) was determined to be approximately **24.6 EFPY**.

Conclusion of ERG P-T Limit Categorization

The Salem Unit 2 reactor vessel material with the highest RT_{NDT} value is the Lower Shell Longitudinal Weld Seams 3-442 A&C. Thus, this material is considered the limiting material for Salem Unit 2. The lower shell plate longitudinal weld seam 3-342 A&C has an EOLR RT_{PTS} (RT_{NDT}) value of 239°F (based on Table 4-1 of Section 4).

The operating time at which the ERG category would transition from Category I to Category II was determined to be approximately **24.6 EFPY**.

9 REFERENCES

1. Code of Federal Regulations, 10 CFR Part 54.3, "Definitions," August 28, 2007.

2. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

3. NUREG-1801, Vol. 2, Rev. 1, Generic Aging Lessons Learned (GALL) Report, U. S. Nuclear Regulatory Commission, Sept. 2005.

4. "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, May 1988.

7. WCAP-15693, "Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," April 2001.

8. PSEG Nuclear LLC Transmittal S-TODI-2008-0010, "Design Clarifications for Upper Shelf Energy, ΔRT_{NDT} and ART, PTS, P-T Limits and Surveillance Capsule Withdrawal Schedule Evaluations and Analyses," September 22, 2008.

9. CMTR-RV-PNJ, "PNJ Reactor Vessel Certified Materials Test Reports."

10. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, Section NB-2300, "Fracture Toughness Requirements for Material".

11. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

12. CE NPSD-1119, Revision 1, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content, CEOG Task 1054," July 1998.

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

14. Code of Federal Regulations, 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 143, dated December 19, 1995, effective January 18, 1996.

15. WCAP-15692, Revision 0, “Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program,” August 2001.

16. Code of Federal Regulations, 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” February 20, 1971.

17. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, “Fracture Toughness Criteria for Protection Against Failure. [Sub-reference AMSE Section XI, Division 1, Code Case N-641, “Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements,” January 17, 2000.]

18. WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

19. Background Information for Westinghouse Owner’s Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 2, April 30, 2005.

APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{IT})

The following pages contain the thermal stress intensity factors (K_{IT}) for the maximum heatup and cooldown rates at 50 EFPY (Tables A-1 and A-2). The vessel radius to the 1/4T and 3/4T locations is as follows:

- 1/4T Radius = 88.875 inches
- 3/4T Radius = 93.188 inches

Table A-1 K_{IT} Values for 100°F/hr. Heatup Curve for 50 EFPY

Water Temp. [°F]	Vessel Temperature @ 1/4T Location for 100°F/hr. Heatup [°F]	1/4T Thermal Stress Intensity Factor [ksi√in]	Vessel Temperature @ 3/4T Location for 100°F/hr. Heatup [°F]	3/4T Thermal Stress Intensity Factor [ksi√in]
60	55.985	-0.9954	55.043	0.4731
65	58.558	-2.4522	55.294	1.4378
70	61.621	-3.7125	55.962	2.4257
75	64.898	-4.9101	57.099	3.3563
80	68.449	-5.9455	58.654	4.1903
85	72.111	-6.8918	60.589	4.9375
90	75.955	-7.7138	62.865	5.5993
95	79.898	-8.4650	65.439	6.1921
100	83.973	-9.1226	68.281	6.7187
105	88.134	-9.7208	71.356	7.1893
110	92.391	-10.2474	74.636	7.6094
115	96.721	-10.7282	78.096	7.9876
120	101.121	-11.1540	81.715	8.3272
125	105.582	-11.5440	85.475	8.6337
130	110.094	-11.8909	89.358	8.9099
135	114.655	-12.2101	93.350	9.1601
140	119.255	-12.4954	97.439	9.3868
145	123.894	-12.7592	101.614	9.5932
150	128.562	-12.9964	105.864	9.7812
155	133.262	-13.2171	110.180	9.9534
160	137.985	-13.4166	114.556	10.1111
165	142.732	-13.6036	118.983	10.2566
170	147.497	-13.7738	123.457	10.3907
175	152.281	-13.9343	127.971	10.5153
180	157.078	-14.0815	132.521	10.6310
185	161.892	-14.2214	137.104	10.7391
190	166.715	-14.3506	141.714	10.8403
195	171.551	-14.4744	146.349	10.9357
200	176.395	-14.5894	151.006	11.0256
205	181.248	-14.7005	155.683	11.1108
210	186.108	-14.8045	160.378	11.1918

Table A-2 K_{IT} Values for 100°F/hr. Cooldown Curve for 50 EFPY

Water Temp. [°F]	Vessel Temperature @ 1/4T Location for 100°F/hr. Cooldown [°F]	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor [ksi√in]
210	237.045	17.1391
205	231.959	17.0704
200	226.873	17.0021
195	221.787	16.9330
190	216.700	16.8642
185	211.613	16.7948
180	206.526	16.7258
175	201.439	16.6562
170	196.352	16.5870
165	191.265	16.5172
160	186.177	16.4480
155	181.089	16.3782
150	176.002	16.3089
145	170.914	16.2393
140	165.827	16.1700
135	160.739	16.1004
130	155.651	16.0313
125	150.564	15.9619
120	145.476	15.8929
115	140.389	15.8237
110	135.302	15.7549
105	130.214	15.6859
100	125.127	15.6173
95	120.040	15.5485
90	114.953	15.4802
85	109.866	15.4116
80	104.779	15.3435
75	99.693	15.2753
70	94.606	15.2075
65	89.520	15.1394
60	84.435	15.0711

Enclosure 4

Westinghouse WCAP-18571-NP, Rev. 2, Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation (Non-Proprietary)

WCAP-18571-NP
Revision 2

July 2022

Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

WCAP-18571-NP
Revision 2

Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

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RECORD OF REVISION

Revision	Description	Completed
0	Original Issue	September 2020
1	This revision corrects on error in Section 2.0 in identifying the cycles used for fluence projections.	August 2021
2	This revision updates Reference 1, WCAP-16982-NP, to Revision 3. The change in referenced revision number did not affect the content of this report. The "Record of Revision" table was also added. Charges are marked with revision bars.	July 2022

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EXECUTIVE SUMMARY

This report provides an evaluation of several parameters for reactor vessel (RV) integrity in order to determine whether the applicable U.S. Nuclear Regulatory Commission (NRC) requirements are met and continued safe operation of Salem Unit 2, with regards to RV integrity, can be justified through end-of-life extension (EOLE), i.e., 60 total years of operation. Specifically, this report documents the following Salem Unit 2 RV integrity calculations/evaluations:

1. Perform an updated neutron fluence assessment for the Salem Unit 2 pressure vessel materials. Cycle-specific analyses for the past and current operating cycles, as well as projections for future operation through 50 effective full-power years (EFPY) are performed.
2. Determine the adjusted reference temperature (ART) values for the RV beltline and extended beltline materials at EOLE, 50 EFPY, i.e., 60 total years of operation for Unit 2.
3. Evaluate the applicability of the pressure-temperature (P-T) limit curves developed in WCAP-16982-NP [Ref. 1], with consideration of updated fluence values. The applicability of the Salem Unit 2 P-T limit curves remains unchanged at 50 EFPY.
4. Determine the pressurized thermal shock (PTS) reference temperature (RT_{PTS}) values for the beltline and extended beltline materials in the RV at 50 EFPY. The RT_{PTS} values of all of the beltline and extended beltline materials in the Salem Unit 2 RV are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds (per 10 CFR 50.61.b.2), through EOLE (50 EFPY).

Appendix A contains a credibility evaluation for surveillance materials considering the updated fluence analysis.

Appendix B contains the evaluation for pressurized thermal shock (PTS) per 10 CFR 50.61 and emergency response guideline (ERG) analysis.

Appendix C contains a validation of the radiation transport calculation model based on neutron dosimetry measurements.

Revision 1 of this WCAP corrects on error in Section 2.0 in identifying the cycles used for fluence projections.

Revision 2 updates Reference 1, WCAP-16982-NP, to Revision 3.

1 INTRODUCTION

The purpose of this report is to evaluate several parameters for reactor vessel (RV) integrity in order to determine whether the applicable U.S. Nuclear Regulatory Commission (NRC) requirements are met and continued safe operation of Salem Unit 2, with regards to RV integrity, can be justified through end-of-life extension (EOLE), i.e., 60 total years of operation.

As part of the time-limited aging analyses (TLAA) performed for the Salem Unit 2 license renewal, a fluence analysis was completed, and new heatup and cooldown P-T limit curves were developed, applicable to 50 EFPY. These analyses are documented in WCAP-16982-NP [Ref. 1].

In this report, a new fluence analysis is performed, using the fully three-dimensional RAPTOR-M3G methodology. Fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV) and iron atom displacements (dpa) are established on a plant- and fuel cycle-specific basis for the first 24 cycles. Based on customer input, projections beyond the current cycle (Cycle 24) are based on Cycles 20 and 23. All of the neutron transport calculations performed in this analysis are based on the nuclear cross-section data derived from ENDF/B-VI and make use of the latest available calculation tools. The neutron transport methodology follows the guidance of Regulatory Guide 1.190 [Ref. 2], and the methods used to determine the RV neutron exposures are consistent with the NRC approved methodology described in WCAP-18124-NP-A [Ref. 3].

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) of the limiting material of the reactor vessel. The adjusted reference temperature (ART) of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} ($RT_{NDT(U)}$) is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, the RT_{NDT} increases associated with the new fluence projections discussed above need to be evaluated to ensure that the limiting RT_{NDT} used in the 50 EFPY P-T limit curves in WCAP-16982-NP remain bounding. To find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The NRC has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [Ref. 4]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)} + \Delta RT_{NDT}$ + margin for uncertainties) at the quarter thickness (1/4T) and three-quarter thickness (3/4T) locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. These results are used to perform an evaluation to confirm the applicability of the P-T curves from WCAP-16982-NP with consideration of updated fluence values.

This report documents the calculated ART values in Section 6. A description of the updated fluence analysis is provided in Section 2 of this report, and a validation of the radiation transport calculation model based on neutron dosimetry measurements is contained in Appendix C.

Appendix A provides a credibility evaluation of the Salem Unit 2 surveillance data. Appendix B contains an evaluation of the Salem Unit 2 pressurized thermal shock (PTS) evaluation at EOLE and emergency response guideline (ERG) analysis.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

Discrete ordinates (S_N) transport analyses were performed to determine the neutron radiation environment within the reactor pressure vessel (RPV). In these analyses, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. The dosimetry analysis documented in Appendix C shows that the ±20% (1σ) acceptance criteria specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. 2] is met, based on the measurement-to-calculation (M/C) comparison results for the in-vessel surveillance capsules withdrawn and analyzed to-date. These validated calculations form the basis for providing projections of the neutron exposure of the RPV through EOLE.

All of the calculations described in this section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET" [Ref. 3]. The neutron transport evaluation methodology described in Reference 3 is based on the guidance of Regulatory Guide 1.190. Note, however, that the NRC Safety Evaluation Report (SER) in Reference 3 states that the applicability of the methodology described in Reference 3 is limited to the traditional RPV beltline region approximated by the RPV region near the active height of the core.

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the RPV, a series of fuel-cycle-specific forward transport calculations were performed using the three-dimensional discrete ordinates code, RAPTOR-M3G [Ref. 3], and the BUGLE-96 cross-section library [Ref. 5]. The BUGLE-96 library provides a coupled 47-neutron and 20-gamma-ray group cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_8 order of angular quadrature. Energy- and space-dependent core power distributions were treated on a fuel-cycle-specific basis.

The Salem Unit 2 reactor is a standard Westinghouse 4-loop design employing reactor internals that include 1.125-inch-thick baffle plates and a fully circumferential thermal shield. The model of the reactor (and reactor cavity) geometry used in the plant-specific evaluation is shown in Figure 2-1 through Figure 2-3.

The model extends radially from the center of the core to 350 cm, azimuthally from 0° to 45° (taking advantage of the octant symmetry of the reactor configuration), and axially from -325 cm to 325 cm with respect to the midplane of the active core. Elevations of key RPV materials relative to the model geometry are provided in Table 2-1.

A plan view of the model geometry at the core midplane is shown in Figure 2-1. In this figure, a single octant is depicted showing the arrangement of the core, reactor internals, core barrel, thermal shield, downcomer, cladding, RPV, reactor cavity, reflective insulation, and bioshield. Depictions of the in-vessel surveillance capsules, including their associated support structures, are also shown.

From a neutronics standpoint, the inclusion of the surveillance capsules and associated support structures in the geometric model is significant. Since the presence of the capsules and support structures has a marked impact on the magnitude of the neutron fluence rate and relative neutron and gamma ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are accounted for in the transport calculations.

A section view of the model geometry is shown in Figure 2-2 and Figure 2-3. Note that the stainless-steel former plates located between the core baffle and barrel regions are shown in these figures.

When developing the reactor model shown in Figure 2-1 through Figure 2-3, nominal design dimensions were employed for the various structural components. Likewise, water temperatures and, hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids and guide tubes.

The geometric mesh description of the reactor model shown in Figure 2-1 through Figure 2-3 consisted of 126 radial by 68 azimuthal by 195 axial intervals. Mesh sizes were chosen to ensure sufficient resolution of the stair-step shaped baffle plates as well as an adequate number of meshes throughout the radial and axial regions of interest. The pointwise inner iteration convergence criterion utilized in the calculations was set at a value of 0.001.

The core power distributions used in the plant-specific transport analysis were taken from nuclear design documentation. The data extracted included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial power distributions were also obtained.

For each fuel cycle of operation, fuel-assembly-specific enrichment and burnup data were used to generate the spatially dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope-dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel-assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the spatial mesh arrays used in the discrete ordinates calculations.

In Table 2-1, axial and azimuthal locations of the RPV materials are provided. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the midplane of the active fuel stack.

Cycle-specific calculations were performed for Cycles 1-24. Note that future fluence projection data beyond Cycle 24 are based on Cycles 20 and 23. At the time of the development of the fluence model, Cycle 24 had not been completed and, thus, the results for Cycle 24 are based on the cycle design data.

Neutron fluence projections for the RPV are given in Table 2-2. Similarly, iron atom displacement results for the RPV are provided in Table 2-3. The data presented represent the maximum neutron exposures experienced by RPV materials. These projections were based on the spatial power distribution and reactor operating conditions of Cycles 20 and 23. The projected results will remain valid as long as future plant operation is consistent with these assumptions. Results of the discrete ordinates transport analyses pertinent to the surveillance capsule evaluations are provided in Appendix C. The reported data also consider both the inner and outer radius of the RPV base metal, and account for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core.

To allow for the determination of potential fast fluence accumulation, the projected fast fluence rate ($E > 1.0$ MeV) at each surveillance capsule location is provided in Table 2-5. Projections of future operation are based on the spatial power distributions and reactor operating conditions of Cycles 20 and 23. Note that RPV neutron exposure rates are dominated by neutron leakage from the peripheral fuel assemblies. The additional fast fluence accumulated for any re-inserted/re-located capsule can be determined by multiplying the fast fluence rate value in Table 2-5 for the appropriate capsule position with the irradiation duration in effective full-power seconds (EFPS).

2.3 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the RPV is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology used in the plant-specific neutron exposure evaluation is carried out in the following four stages:

1. Comparisons of calculations with benchmark measurements from the pool critical assembly (PCA) simulator (NUREG/CR-6454, “Pool Critical Assembly Pressure Vessel Facility Benchmark” [Ref. 6]) at the Oak Ridge National Laboratory (ORNL) and the VENUS-1 experiment.
2. Comparison of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment (NUREG/CR-6453, “H.B. Robinson-2 Pressure Vessel Benchmark” [Ref. 7]).
3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments (WCAP-18124-NP-A [Ref. 3]).
4. Comparison of the calculations with all available dosimetry results from the RPV measurement programs carried out at Salem Unit 2 (Appendix C).

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This phase, however, did not test the accuracy of commercial core neutron source calculations, nor did it address uncertainties in operational and geometric variables that impact power reactor calculations.

The second phase of the qualification (H.B. Robinson comparisons) addressed uncertainties that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations.

The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational method approximations as well as to a lack of knowledge relative to various plant-specific parameters. The overall calculational uncertainty applicable to the Salem Unit 2 analyses were established from the results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons of plant-specific dosimetry measurements) was used solely to demonstrate the adequacy of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used to bias the final results in any way.

Table 2-6 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in WCAP-18124-NP-A. The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons given in Table 2-4 support these uncertainty assessments for Salem Unit 2.

Table 2-1
RPV Material Locations

Material	Azimuthal Location[a] [°] Min	Azimuthal Location[a] [°] Max	Axial Location[b] [cm] Min	Axial Location[b] [cm] Max
Outlet nozzle to nozzle shell welds				
- Nozzle 1	22.0		270.12	
- Nozzle 2	158.0		270.12	
- Nozzle 3	202.0		270.12	
- Nozzle 4	338.0		270.12	
Inlet nozzle to nozzle shell welds				
- Nozzle 1	67.0		265.04	
- Nozzle 2	113.0		265.04	
- Nozzle 3	247.0		265.04	
- Nozzle 4	293.0		265.04	
Nozzle shell				
- Plate 1	300.0	60.0	233.53	485.07
- Plate 2	60.0	180.0	233.53	485.07
- Plate 3	180.0	300.0	233.53	485.07
Nozzle shell longitudinal welds				
- Weld 1	60.0		233.53	485.07
- Weld 2	180.0		233.53	485.07
- Weld 3	300.0		233.53	485.07
Nozzle shell to intermediate shell circumferential weld – centerline	0.0	360.0	233.53	
Intermediate shell				
- Plate 1	0.0	120.0	-42.93	233.53
- Plate 2	120.0	240.0	-42.93	233.53
- Plate 3	240.0	360.0	-42.93	233.53
Intermediate shell longitudinal welds				
- Weld 1	0.0		-42.93	233.53
- Weld 2	120.0		-42.93	233.53
- Weld 3	240.0		-42.93	233.53
Intermediate shell to lower shell circumferential weld – centerline	0.0	360.0	-42.93	
Lower shell				
- Plate 1	300.0	60.0	-313.52	-42.93
- Plate 2	60.0	180.0	-313.52	-42.93
- Plate 3	180.0	300.0	-313.52	-42.93
Lower shell longitudinal welds				
- Weld 1	60.0		-313.52	-42.93
- Weld 2	180.0		-313.52	-42.93
- Weld 3	300.0		-313.52	-42.93
Lower shell to lower vessel head circumferential weld – centerline	0.0	360.0	-313.52	

Notes:
(a) Azimuthal locations are indexed to $\theta = 0.0$.
(b) Axial elevations are indexed to $Z = 0.0$ at the midplane of the active fuel stack.

Table 2-2
Projection of Maximum Fast Neutron (E > 1.0 MeV) Fluence
Experienced by Reactor Pressure Vessel Materials in the Beltline

Beltline Material	Neutron Fluence (n/cm^2)		
	32 EFPY	48 EFPY	50 EFPY
Vessel shells			
Lower shell	1.34E+19	1.93E+19	2.01E+19
Intermediate shell	1.36E+19	1.97E+19	2.05E+19
Upper shell	1.71E+17	2.36E+17	2.44E+17
Circumferential (or girth) welds			
Bottom torus/lower shell	2.21E+15	3.12E+15	3.24E+15
Lower shell/intermediate shell	1.32E+19	1.91E+19	1.98E+19
Intermediate shell/upper shell	1.96E+17	2.71E+17	2.80E+17
Longitudinal welds			
Lower shell			
60° and 300°	9.56E+18	1.39E+19	1.45E+19
180°	5.00E+18	7.11E+18	7.38E+18
Intermediate shell			
0°	4.97E+18	7.14E+18	7.41E+18
120° and 240°	9.61E+18	1.41E+19	1.47E+19
Upper shell			
60° and 300°	1.27E+17	1.77E+17	1.83E+17
180°	7.10E+16	9.77E+16	1.01E+17
Nozzle forging attachment weld			
Inlet nozzle	1.56E+16	2.18E+16	2.26E+16
Outlet nozzle	1.23E+16	1.72E+16	1.78E+16

Table 2-3
Projection of Iron Atom Displacements (dpa)
Experienced by Reactor Pressure Vessel Materials in the Beltline

Beltline Material	Iron Atom Displacements (dpa)		
	32 EFPY	48 EFPY	50 EFPY
Vessel shells			
Lower shell	2.16E-02	3.11E-02	3.23E-02
Intermediate shell	2.20E-02	3.19E-02	3.31E-02
Upper shell	3.10E-04	4.29E-04	4.44E-04
Circumferential (or girth) welds			
Bottom torus/lower shell	1.40E-05	1.97E-05	2.04E-05
Lower shell/intermediate shell	2.14E-02	3.10E-02	3.21E-02
Intermediate shell/upper shell	3.55E-04	4.90E-04	5.07E-04
Longitudinal welds			
Lower shell			
60° and 300°	1.54E-02	2.24E-02	2.33E-02
180°	8.02E-03	1.14E-02	1.19E-02
Intermediate shell			
0°	8.06E-03	1.16E-02	1.20E-02
120° and 240°	1.55E-02	2.28E-02	2.37E-02
Upper shell			
60° and 300°	2.34E-04	3.25E-04	3.37E-04
180°	1.31E-04	1.81E-04	1.87E-04
Nozzle forging attachment weld			
Inlet nozzle	4.15E-05	5.89E-05	6.10E-05
Outlet nozzle	3.44E-05	4.88E-05	5.06E-05

Table 2-4
Surveillance Capsule Exposure Values

Capsule	Capsule Location (First-Octant-Equivalent)	Withdrawn (EOC)	Withdrawal (EFPY)	Capsule Fluence (E > 1.0 MeV) (n/cm^2)	Lead Factor
T	40° (40°)	1	1.2	2.73E+18	3.31
U	140° (40°)	3	2.7	5.81E+18	3.35
X	220° (40°)	6	6.2	1.13E+19	3.36
Y	320° (40°)	11	10.8	1.83E+19	3.36

Table 2-5
Projection of Surveillance Capsule Fluence and Lead Factors

	Capsule Fluence (E > 1.0 MeV) (n/cm^2)		Lead Factor	
EFPY	4° Capsule	40° Capsule	4° Capsule	40° Capsule
32	1.57E+19	4.57E+19	1.15	3.36
48	2.25E+19	6.62E+19	1.14	3.36
50	2.34E+19	6.88E+19	1.14	3.36

Table 2-6
Calculational Uncertainties

Description	Uncertainty	
	Capsule	Vessel Inner Radius
PCA comparisons	3%	3%
H.B. Robinson comparisons	5%	5%
Analytical sensitivity studies	9%	11%
Additional uncertainty for factors not explicitly evaluated	5%	5%
Net calculational uncertainty	12%	13%

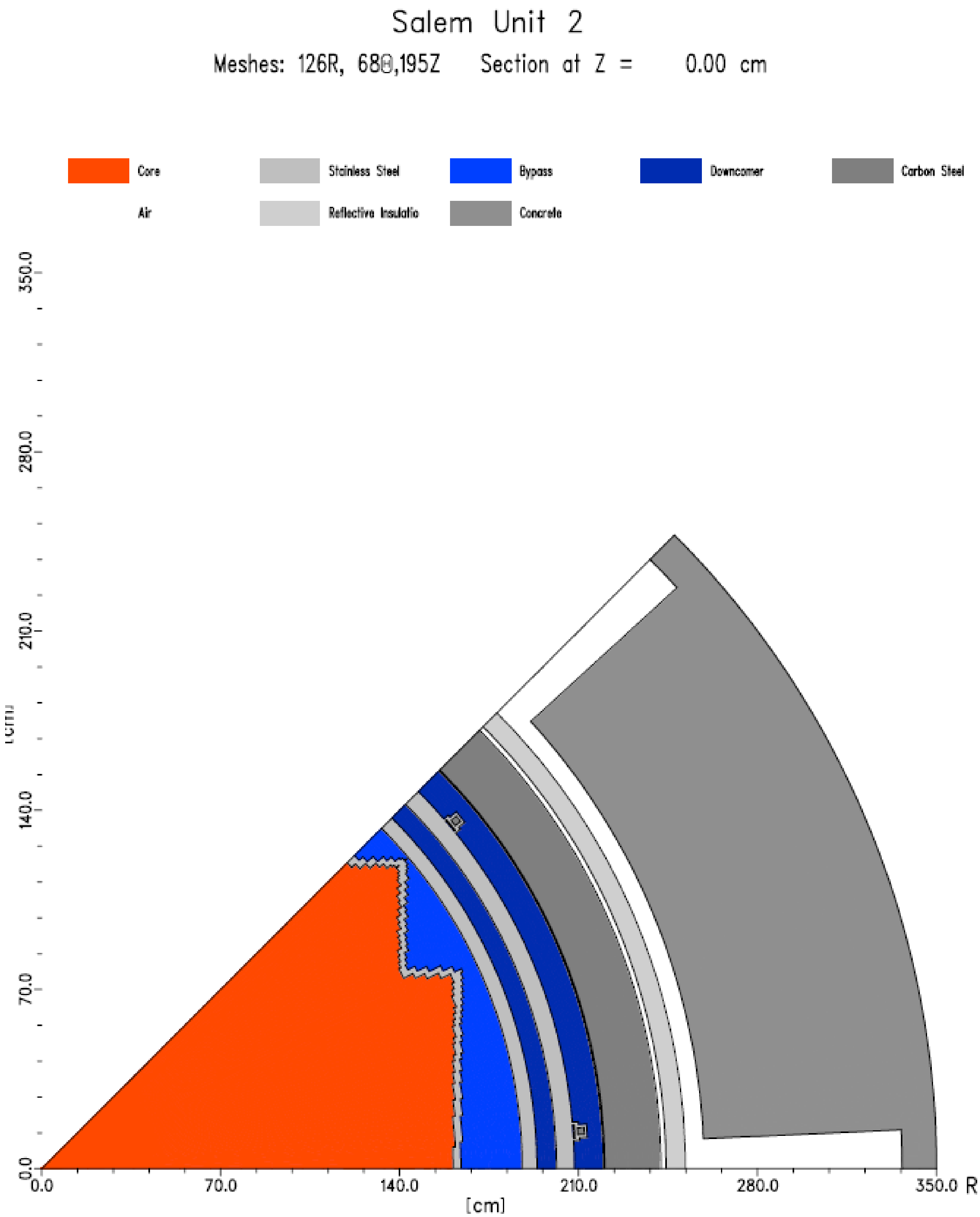

Figure 2-1 Plan View of the Reactor Geometry at the Core Midplane

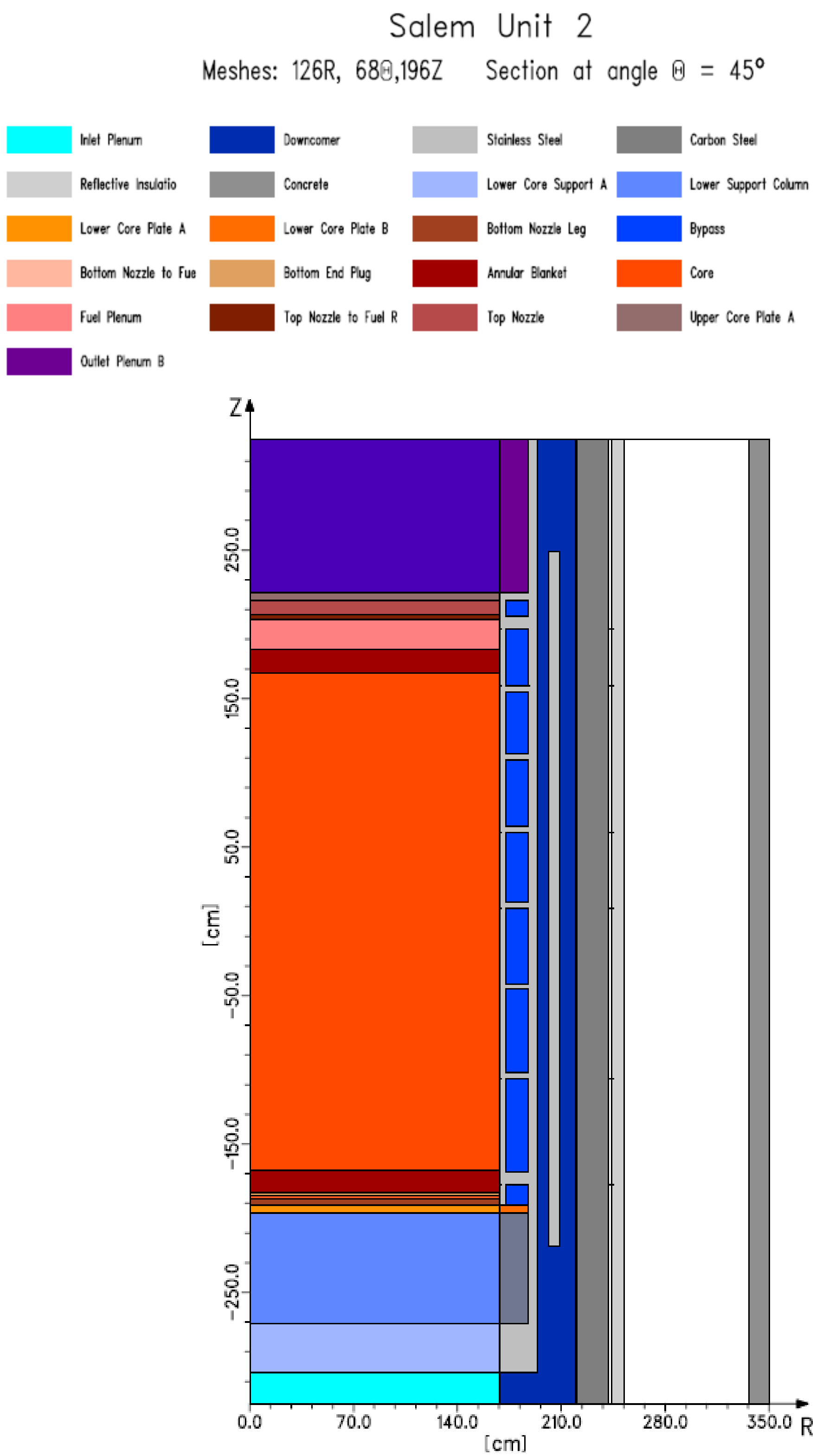

Figure 2-2 Section View of the Reactor Geometry – 45° Azimuth

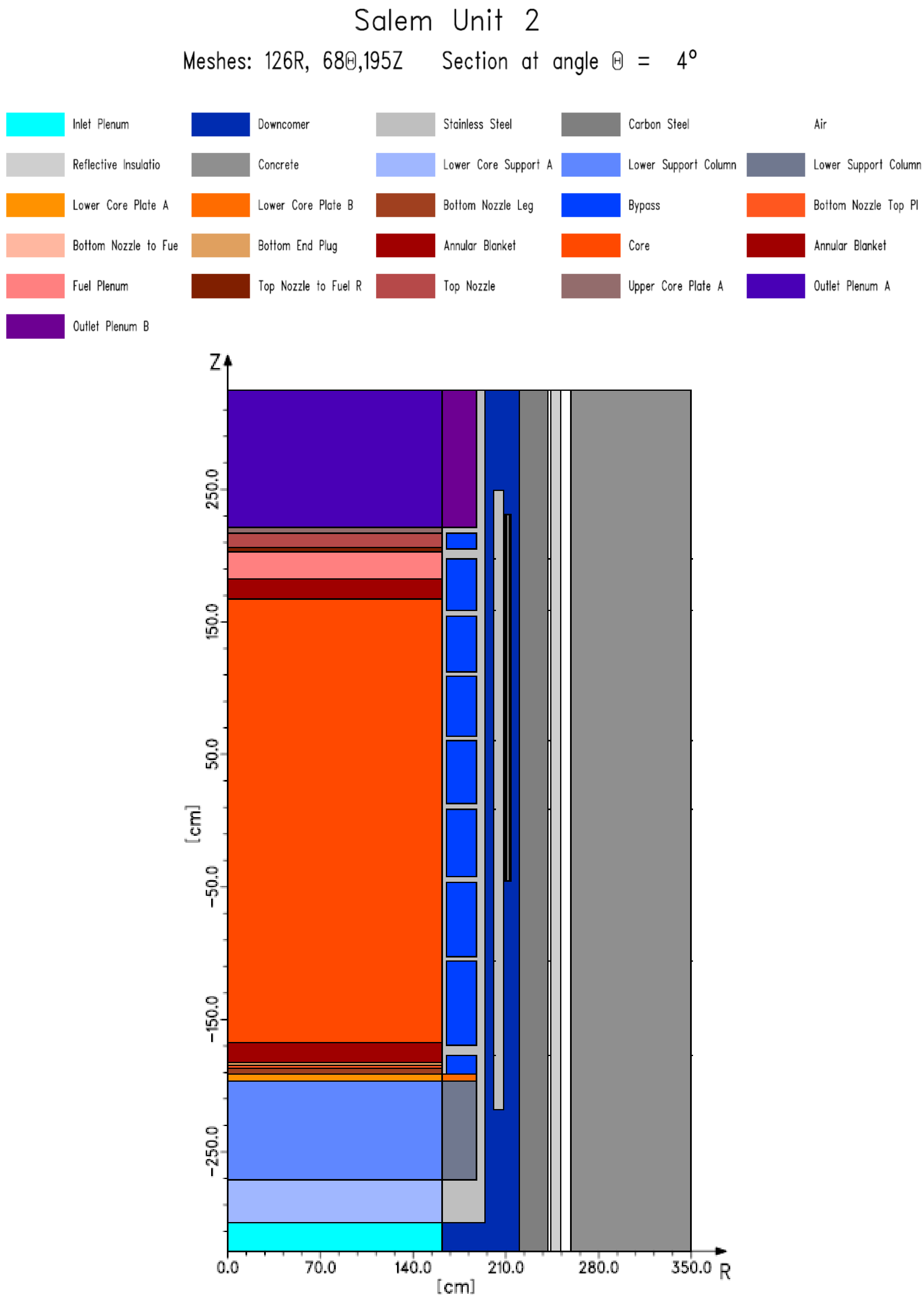

Figure 2-3 Section View of the Reactor Geometry – 4° Azimuth

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [Ref. 8]. The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

> *…the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

The Salem Unit 2 beltline materials traditionally included the intermediate and lower shell plates and welds; however, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. 9], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves. The additional materials that exceed this fluence threshold are referred to as the "extended beltline" materials and are evaluated to ensure that the applicable neutron embrittlement effects are considered. As seen from Table 2-2 of this report, the extended beltline materials include upper shell plates, upper shell longitudinal welds, and the upper to intermediate shell girth weld. The fluence for both the inlet and outlet nozzle to upper shell welds are less than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at 50 EFPY. Therefore, the materials of the inlet/outlet nozzle forgings and the associated welds to the upper shell do not need to be considered in the extended beltline. Note that for reactor vessel welds, the terms "girth" and "circumferential" are used interchangeably; herein, these welds shall be referred to as circumferential welds. Similarly, for reactor vessel welds, the terms "axial" and "longitudinal" are used interchangeably; herein, these welds shall be referred to as longitudinal welds.

Although the reactor vessel nozzles are not a part of the extended beltline, per NRC RIS 2014-11, the nozzle materials must be evaluated for their potential effect on P-T limit curves due to the higher stresses in the nozzle corner region. These higher stresses can potentially result in more restrictive P-T limits, even if the RT_{NDT} for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries. The effect of these higher stresses is addressed in Section 7, which determines that the Salem Unit 2 beltline P-T limit curves generated in WCAP-16982-NP [Ref. 1] bound the inlet and outlet nozzle P -T limit curves.

A summary of the best-estimate copper (Cu) and nickel (Ni) contents in units of weight percent (wt. %), as well as initial RT_{NDT} values, for the Salem Unit 2 reactor vessel beltline and extended beltline materials are provided in Table 3-1. Table 3-2 provides the best-estimate Cu and Ni wt. % values and initial RT_{NDT} values for the reactor vessel nozzle forgings.

Table 3-1 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Salem Unit 2 Reactor Vessel Materials[a]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ (°F)	σ_I[b] (°F)
Upper Shell B4711-1	C4194-1	---	0.11	0.55	60	0
Upper Shell B4711-2	C4149-2	---	0.14	0.56	60	0
Upper Shell B4711-3	C4171-1	---	0.12	0.58	101	0
Intermediate Shell B4712-1	C4173-1	---	0.13	0.56	0	0
Intermediate Shell B4712-2	C4186-2	---	0.12	0.61	12	0
Intermediate Shell B4712-3	C4194-2	---	0.11	0.57	10	0
Lower Shell B4713-1	C4182-1	---	0.12	0.60	8	0
Lower Shell B4713-2	C4182-2	---	0.12	0.57	8	0
Lower Shell B4713-3	B-8343-1	---	0.12	0.58	10	0
Intermediate to Upper Shell Circumferential Weld Seam 8-442	20291/1P2809	Linde 1092 (Lot # 3854)	0.27[c]	0.735[c]	-56[d]	17[d]
Intermediate to Lower Shell Circumferential Weld Seam 9-442	90099	Linde 0091 (Lot # 3977)	0.197	0.06	-56[d]	17[d]
Upper Shell Longitudinal Weld Seams 1-442 A, B, & C	13253/12008	Linde 1092 (Lot # 3833)	0.21[c]	0.873[c]	-56[d]	17[d]
Intermediate Shell Longitudinal Weld Seams 2-442 A, B, & C	13253/20291	Linde 1092 (Lot # 3833)	0.221	0.732	-56[d]	17[d]
Lower Shell Longitudinal Weld Seams 3-442 A, B, & C	21935/12008	Linde 1092 (Lot # 3889)	0.213	0.867	-56[d]	17[d]
Salem Unit 2 Surveillance Weld Material[f]	13253	Linde 1092 (Lot # 3883/3774)	0.225	0.727	---	---
Diablo Canyon Unit 2 Surveillance Program Weld Metal	21935/12008[e]	Linde 1092 (Lot # 3869)[e]	0.22[e]	0.87[e]	---	---

Notes:

(a) The data was extracted from WCAP-16982-NP [Ref. 1], unless otherwise noted.

(b) All $RT_{NDT(U)}$ values are based on measured data with a σ_I = 0°F, unless otherwise noted.

(c) Best estimate copper and nickel content based on CE NPSD-1119, Revision 1 [Ref. 10]. Note that in the case of the Intermediate to Upper Shell Circumferential Weld Seam 8-442, there is no record that the two Heat Numbers 20291 and 1P2809 were deposited in tandem; therefore, the copper and nickel content for Heat # 1P2809 is used as a more conservative basis for estimating the effect of neutron irradiation.

(d) This value is a generic weld $RT_{NDT(U)}$ value from 10 CFR 50.61 [Ref. 11]. The use of this estimated $RT_{NDT(U)}$ requires σ_I = 17°F.

(e) The Diablo Canyon Unit 2 surveillance information is taken from WCAP-15423 [Ref. 12].

(f) Note that the Salem Unit 2 surveillance weld metal Heat # 13253 is only representative of the beltline welds, not identical. Therefore, the surveillance weld data was not used in the calculations documented in this report.

Table 3-2 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the Salem Unit 2 Reactor Vessel Inlet/Outlet Nozzle Materials[(a)]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	$RT_{NDT(U)}$ (°F)	σ_I[(b)] (°F)
Inlet Nozzle B4703-1	ZV-3265	---	N/A	0.69	-27	0
Inlet Nozzle B4703-2	ZV-3265	---	N/A	0.685	-29	0
Inlet Nozzle B4703-3	ZV-3265	---	N/A	0.675	-23	0
Inlet Nozzle B4703-4	SV 2040-1	---	N/A	0.805	-31	0
Outlet Nozzle B4704-1	AV-2042	---	N/A	0.835	-68	0
Outlet Nozzle B4704-2	AV-2061	---	N/A	0.77	-44	0
Outlet Nozzle B4704-3	AV-2067	---	N/A	0.69	-72	0
Outlet Nozzle B4704-4	AV-2099	---	N/A	0.71	-51	0

Notes:

(a) The inlet/outlet nozzle forgings are projected to be less than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at EOLE and do not need to be considered for the effects of embrittlement. However, the initial material properties are provided here for future reference. Note the initial RT_{NDT} values were calculated, consistent with the Unit 1 nozzles in WCAP-18502-NP [Ref. 13], using the methodology in BWRVIP-173-A [Ref. 14] to supplement the ASME Section III methodology to account for limited material testing data available.

(b) All $RT_{NDT(U)}$ values are based on measured data with σ_I = 0°F.

4 SURVEILLANCE DATA

Per Regulatory Guide 1.99, Revision 2 [Ref. 4], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, data from surveillance programs at other plants, which include a Salem Unit 2 reactor vessel beltline or extended beltline materials, should also be considered when calculating Position 2.1 chemistry factors. Data from a surveillance program at another plant is often called 'sister-plant' data.

The Salem Unit 2 surveillance capsules contain plate material from Intermediate Shell Plate B4712-2. Table 4-1 summarizes the surveillance data available for the Salem Unit 2 plate materials that will be used in the calculation of the Position 2.1 chemistry factor values. Per Appendix A, the surveillance data are deemed credible for Salem Unit 2.

The Salem Unit 2 surveillance weld specimens were fabricated from weld wire Heat # 13253. Table 4-1 summarizes the surveillance data available for the Salem Unit 2 weld materials. However, the Salem Unit 2 surveillance weld wire Heat # 13253 is only representative of the beltline welds, not identical. Therefore, the surveillance weld data are not used, and the data are provided in Table 4-1 for information only.

The Diablo Canyon Unit 2 surveillance program contains weld wire Heat # 21935/12008 with Linde 1092 flux, which was also used in the fabrication of Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442 A, B, & C. Thus, the data from this surveillance program is applicable to Salem Unit 2. Table 4-2 contains the sister-plant weld material surveillance data. Per WCAP-17315-NP [Ref. 15], the surveillance data are deemed credible; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 6 for the Salem Unit 2 lower shell longitudinal weld seams.

Table 4-1 Salem Unit 2 Surveillance Capsule Data[(a)]

Material	Capsule	Capsule Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured ΔRT_{NDT} (°F)	Irradiation Temperature (°F)
Intermediate Shell Plate B4712-2 (longitudinal)	T	0.273	61.66	545
	U	0.581	66.54	543
	X	1.13	93.82	541
	Y	1.83	105.69	540
Intermediate Shell Plate B4712-2 (transverse)	T	0.273	74.83	545
	U	0.581	98.26	543
	X	1.13	125.15	541
	Y	1.83	129.33	540
Salem Unit 2 Surveillance Weld Material[(c)] (Heat #13253)	T	0.273	153.17	545
	U	0.581	185.94	543
	X	1.13	195.43	541
	Y	1.83	200.90	540

Notes:

(a) Information extracted from WCAP-15692 [Ref. 16], unless otherwise noted.
(b) The fluence values are taken from Table 2-4.
(c) Note that the Salem Unit 2 surveillance weld metal Heat # 13253 is only representative of the beltline welds, not identical. Therefore, the surveillance weld daata was not used in the calculations documented in this report.

Table 4-2 Sister-Plant Surveillance Program Results

Material	Capsule	Withdrawal		Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured ΔRT_{NDT} (°F)	Irradiation Temperature (°F)
		Cycle	EFPY			
Diablo Canyon Unit 2 Surveillance Weld Material[(a)] (Heat # 21935/12008)	U	1	1.02	0.330	173.0	545
	X	3	3.16	0.906	203.2	545
	Y	6	7.08	1.53	211.4	545
	V	9	11.49	2.38	224.5	545

Note:

(a) The Diablo Canyon Unit 2 capsule data were taken from WCAP-17315-NP [Ref. 15]. EFPY values are found in WCAP-15423 [Ref. 12].

5 CHEMISTRY FACTORS

The chemistry factors (CFs) are calculated using Regulatory Guide 1.99, Revision 2 [Ref. 4], Positions 1.1 and 2.1. Position 1.1 chemistry factors for each reactor vessel material are calculated using the best-estimate copper and nickel weight percent of the material and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2. The best-estimate copper and nickel weight percent values for the Salem Unit 2 reactor vessel materials are provided in Table 3-1 of this report. Regulatory Guide 1.99, Position 1.1 chemistry factors are listed in Table 5-3.

The Position 2.1 CFs are calculated for the materials that have available surveillance data from the plant-specific surveillance program. The Position 2.1 CF calculation is performed using the method described in Regulatory Guide 1.99, Revision 2. The Salem Unit 2 surveillance data are summarized in Section 4 of this report and will be utilized in the Position 2.1 CF calculations in this section. The Position 2.1 CF calculations are presented in Table 5-1 for the Salem Unit 2 surveillance materials. The Position 2.1 CF calculations are presented in Table 5-2 for the sister-plant surveillance material.

In addition to the plant-specific surveillance data, data from surveillance programs at other plants which include a Salem Unit 2 reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 CFs. As discussed in Section 4, Salem Unit 2 does utilize surveillance data from a sister plant in the beltline or extended beltline. Adjustment of the ΔRT_{NDT} values are required per Regulatory Guide 1.99, Revision 2 due to chemistry differences between the surveillance welds and the Salem Unit 2 reactor vessel welds. The chemistry adjustment factor based on the differences between the Regulatory Guide 1.99, Position 1.1 CFs are determined below.

<u>Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442A, 3-442B, and 3-442C (Heat # 21935/12008)</u>

Diablo Canyon Unit 2 data
$CF_{\text{Beltline Weld (Salem 2)}} = 208.6°F$
$CF_{\text{Surv. Weld (Diablo Canyon 2)}} = 211.2°F$
Ratio = 208.6 / 211.2 = 0.99

The ratio procedure results in a ratio of less than 1.0; therefore, the chemistry adjustment can conservatively be neglected.

From NRC Industry Meetings on November 12, 1997 and February 12 and 14, 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one reactor vessel applied to another reactor vessel. The following is taken from the handout [Ref. 17] given by the NRC at those industry meetings.

Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.

Thus, for plants that use surveillance data from other reactor vessels that operate at a different temperature, or when the capsule is at a different temperature than the plant, then this difference must be considered. The temperature adjustment procedure is not applied when considering only surveillance data from the plant-specific program. The temperature adjustment is as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\ Measured} + (T_{capsule} - T_{Plant}) \quad (1)$$

When the surveillance data from capsules irradiated in other plants are applied to the Salem Unit 2 reactor vessel welds, the data must be adjusted to the Salem Unit 2 temperature, T_{cold}, using the equation listed above. The time-averaged irradiation temperature of the Salem Unit 2 vessel over the lifetime of the plant is 542°F. The irradiated temperatures of the surveillance capsules from Diablo Canyon Unit 2 are provided in Table 4-2.

The Position 1.1 and Position 2.1 chemistry factors are summarized in Table 5-3 for Salem Unit 2.

Table 5-1 Calculation of Chemistry Factors Using Salem Unit 2 Surveillance Capsule Data[(a)]

Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT} (°F)	FF* ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate B4712-2 (longitudinal)	T	0.273	0.646	61.66	39.84	0.418
	U	0.581	0.848	66.54	56.43	0.719
	X	1.13	1.034	93.82	97.02	1.069
	Y	1.83	1.166	105.69	123.21	1.359
Intermediate Shell Plate B4712-2 (transverse)	T	0.273	0.646	74.83	48.35	0.418
	U	0.581	0.848	98.26	83.33	0.719
	X	1.13	1.034	125.15	129.42	1.069
	Y	1.83	1.166	129.33	150.76	1.359
	SUM:				728.36	7.130
	$CF_{B4712-2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (728.36) \div (7.130) = 102.2°F$					
Salem Unit 2 Surveillance Weld Material (Heat # 13253)	T	0.273	0.646	153.17	98.97	0.418
	U	0.581	0.848	185.94	157.68	0.719
	X	1.13	1.034	195.43	202.10	1.069
	Y	1.83	1.166	200.90	234.20	1.359
	SUM:				692.95	3.565
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (692.95) \div (3.565) = 194.4°F$					

Notes:

(a) Fluence and measured ΔRT_{NDT} are taken from Table 4-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10l*\log(f))}$.

Table 5-2 Calculation of Chemistry Factors Using Diablo Canyon Unit 2 Surveillance Capsule Data[(a)]

Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Adjusted ΔRT_{NDT}[(c)] (°F)	FF* ΔRT_{NDT} (°F)	FF^2
Diablo Canyon Unit 2 Surveillance Weld Material (Heat # 21935/12008)	U	0.330	0.695	176.0 (173.0)	122.32	0.483
	X	0.906	0.972	206.2 (203.2)	200.49	0.945
	Y	1.53	1.118	214.4 (211.4)	239.62	1.249
	V	2.38	1.234	227.5 (224.5)	280.70	1.522
	SUM:				843.14	4.200
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (843.14) \div (4.200) = 200.7°F$					

Notes:

(a) Fluence and measured ΔRT_{NDT} taken from Table 4-2.

(b) FF = fluence factor = $f^{(0.28 - 0.10 \log(f))}$.

(c) The surveillance weld ΔRT_{NDT} values have been adjusted for irradiation temperature as follows:
Adjusted $\Delta RT_{NDT} = \Delta RT_{NDT,\ Measured}$ + Temp. Adjustment
The temperature adjustments are based on a Salem Unit 2 reactor vessel temperature of 542°F. The Diablo Canyon Unit 2 capsule irradiation temperatures are provided in Table 4-2. The measured (unadjusted) ΔRT_{NDT} values are shown in parenthesis. The temp. adjustment is therefore 3°F (545°F - 542°F).

Table 5-3 Position 1.1 and 2.1 Chemistry Factors for Salem Unit 2

Material	Chemistry Factor	
	Position 1.1[(a)] (°F)	Position 2.1[(b)] (°F)
Upper Shell B4711-1	73.5	---
Upper Shell B4711-2	98.2	---
Upper Shell B4711-3	82.6	---
Intermediate Shell B4712-1	89.8	---
Intermediate Shell B4712-2	83.2	102.2
Intermediate Shell B4712-3	73.7	---
Lower Shell B4713-1	83.0	---
Lower Shell B4713-2	82.4	---
Lower Shell B4713-3	82.6	---
Intermediate to Upper Shell Circumferential Weld Seam 8-442	205.6	---
Intermediate to Lower Shell Circumferential Weld Seam 9-442	91.4	---
Upper Shell Longitudinal Weld Seams 1-442 A, B, & C	208.7	---
Intermediate Shell Longitudinal Weld Seams 2-442 A, B, & C	189.1	---
Lower Shell Longitudinal Weld Seams 3-442 A, B, & C	208.6	200.7
Salem Unit 2 Surveillance Weld Material	189.3	---
Diablo Canyon Unit 2 Surveillance Program Weld Metal	211.2	---

Notes:

(a) All values are basd on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [Ref. 4] (Position 1.1) and the Cu and Ni weight percent values given in Table 3-1.

(b) Values are from Table 5-1 or Table 5-2.

6 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [Ref. 4], the ART for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (2)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code [Ref. 18]. If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log (f))} \quad (3)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth:

$$f_{(depth\ x)} = f_{surf} * e^{(-0.24x)} \quad (4)$$

Where:

x = depth into the vessel wall measured from the vessel clad/base metal interface in inches (reactor vessel cylindrical shell beltline thickness is 8.625 inches).

The resultant fluence is then placed in Equation 3 to calculate the ΔRT_{NDT} at the specific depth. The projected reactor vessel neutron fluence was updated for this analysis and is documented in Section 2 of this report.

Table 6-1 contains the surface fluence values at 50 EFPY, as well as the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2. The values in this table will be used to calculate the EOLE ART values for the Salem Unit 2 reactor vessel materials.

Margin is calculated as $M = 2\sqrt{\sigma_I{}^2 + \sigma_\Delta{}^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is 0°F when the initial RT_{NDT} is a measured value. When a generic value is used, the σ_I is obtained from the set of data used to establish the mean. The standard deviation for the ΔRT_{NDT} margin term (σ_Δ) is 17°F for plates or forgings when surveillance data are not used or is non-credible, and 8.5°F (half the value) for plates or forgings when credible surveillance data are used. For welds, σ_Δ is equal to 28°F when surveillance capsule data are not used or is non-credible and is 14°F (half the value) when credible surveillance capsule data are used. The value for σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Per Appendix A, the surveillance data for Intermediate Shell Plate B4712-2 are deemed credible. The surveillance weld data for Heat # 21935/12008 from the Diablo Canyon Unit 2 surveillance program are also deemed credible.

Tables 6-2 and 6-3 contain the 50 EFPY ART calculations at the 1/4T and 3/4T locations, respectively. The limiting ART values are summarized in Table 6-4.

Table 6-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T and 3/4T Locations for the Salem Unit 2 Reactor Vessel Materials at 50 EFPY

Material	Surface Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF[(c)]	1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(c)]	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(c)]
Upper shell plates	0.0244	0.194	0.0145	0.141	0.00517	0.069
Intermediate shell plates	2.05	1.196	1.22	1.056	0.434	0.768
Lower shell plates	2.01	1.190	1.20	1.050	0.426	0.763
Intermediate to Upper Shell Circumferential Weld Seam 8-442	0.028	0.211	0.0167	0.154	0.00593	0.076
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.98	1.186	1.18	1.046	0.419	0.759
Upper Shell Longitudinal Weld Seams 1-442 A & C	0.0183	0.163	0.0109	0.116	0.00387	0.055
Upper Shell Longitudinal Weld Seam 1-442 B	0.0101	0.110	0.0060	0.077	0.00214	0.035
Intermediate Shell Longitudinal Weld Seams 2-442 B & C	1.47	1.107	0.876	0.963	0.311	0.680
Intermediate Shell Longitudinal Weld Seam 2-442 A	0.741	0.916	0.442	0.773	0.157	0.513
Lower Shell Longitudinal Weld Seams 3-442 A & C	1.45	1.103	0.864	0.959	0.307	0.676
Lower Shell Longitudinal Weld Seam 3-442 B	0.738	0.915	0.440	0.772	0.156	0.512

Notes:

(a) Fluence values are documented in Table 2-2.

(b) The 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (8.625 inches), and equation $f = f_{surf} * e^{-0.24(x)}$ from Regulatory Guide 1.99, Revision 2, where x = depth into the vessel wall (inches).

(c) FF = fluence factor = $f^{(0.28 - 0.10 \log(f))}$.

Table 6-2 Adjusted Reference Temperature Evaluation for the Salem Unit 2 Reactor Vessel Materials Through 50 EFPY at the 1/4T Location[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	1/4T Fluence[(c)] (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Upper Shell B4711-1	1.1	73.5	0.0145	0.141	60	10.3	0.0	5.2	10.3	80.7
Upper Shell B4711-2	1.1	98.2	0.0145	0.141	60	13.8	0.0	6.9	13.8	87.6
Upper Shell B4711-3	1.1	82.6	0.0145	0.141	101	11.6	0.0	5.8	11.6	124.2
Intermediate Shell B4712-1	1.1	89.8	1.22	1.056	0	94.8	0.0	17.0	34.0	128.8
Intermediate Shell B4712-2	1.1	83.2	1.22	1.056	12	87.8	0.0	17.0	34.0	133.8
Using Credible Surveillance Data[(f)]	2.1	102.2	1.22	1.056	12	107.9	0.0	8.5	17.0	136.9[(f)]
Intermediate Shell B4712-3	1.1	73.7	1.22	1.056	10	77.8	0.0	17.0	34.0	121.8
Lower Shell B4713-1	1.1	83.0	1.20	1.050	8	87.2	0.0	17.0	34.0	129.2
Lower Shell B4713-2	1.1	82.4	1.20	1.050	8	86.6	0.0	17.0	34.0	128.6
Lower Shell B4713-3	1.1	82.6	1.20	1.050	10	86.8	0.0	17.0	34.0	130.8
Intermediate to Upper Shell Circumferential Weld Seam 8-442	1.1	205.6	0.0167	0.154	-56	31.6	17.0	15.8	46.4	22.0
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.1	91.4	1.18	1.046	-56	95.6	17.0	28.0	65.5	105.1
Upper Shell Longitudinal Weld Seams 1-442 A & C (60° / 300°)	1.1	208.7	0.0109	0.116	-56	24.3	17.0	12.1	41.8	10.0
Upper Shell Longitudinal Weld Seam 1-442 B (180°)	1.1	208.7	0.00602	0.0768	-56	16.0	17.0	8.0	37.6	-2.4
Intermediate Shell Longitudinal Weld Seams 2-442 B & C (120° / 240°)	1.1	189.1	0.876	0.963	-56	182.1	17.0	28.0	65.5	191.6
Intermediate Shell Longitudinal Weld Seam 2-442 A (0°)	1.1	189.1	0.442	0.773	-56	146.1	17.0	28.0	65.5	155.6
Lower Shell Longitudinal Weld Seams 3-442 A & C (60° / 300°)	1.1	208.6	0.864	0.959	-56	200.1	17.0	28.0	65.5	209.6
Using Credible Surveillance Data for Weld Heat # 21935/12008[(f)]	2.1	200.7	0.864	0.959	-56	192.5	17.0	14.0	44.0	180.5[(f)]
Lower Shell Longitudinal Weld Seam 3-442 B (180°)	1.1	208.6	0.440	0.772	-56	161.0	17.0	28.0	65.5	170.5
Using Credible Surveillance Data for Weld Heat # 21935/12008[(f)]	2.1	200.7	0.440	0.772	-56	154.9	17.0	14.0	44.0	142.9[(f)]

Notes for Table 6-2 contained on the next page.

Notes for Table 6-2:

(a) The Regulatory Guide 1.99, Revision 2 [Ref. 4] methodology was utilized in the calculation of the ART values.

(b) CFs are taken from Table 5-3.

(c) Fluence and FFs are taken from Table 6-1.

(d) $RT_{NDT(U)}$ and σ_I values are taken from Table 3-1.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(f) The credibility evaluation in Appendix A of this report determined that the surveillance data for Intermediate Shell Plate B4712-2 are deemed credible. The surveillance weld data for Heat # 21935/12008 from the Diablo Canyon Unit 2 surveillance program, applicable to the Lower Shell Longitudinal Weld Seams 3-442 A, B, & C are deemed credible in WCAP-17315-NP [Ref. 15]. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF.

Table 6-3 Adjusted Reference Temperature Evaluation for the Salem Unit 2 Reactor Vessel Materials Through 50 EFPY at the 3/4T Location[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	3/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	M (°F)	ART (°F)
Upper Shell B4711-1	1.1	73.5	0.00517	0.069	60	5.0	0.0	2.5	5.0	70.1
Upper Shell B4711-2	1.1	98.2	0.00517	0.069	60	6.7	0.0	3.4	6.7	73.5
Upper Shell B4711-3	1.1	82.6	0.00517	0.069	101	5.7	0.0	2.8	5.7	112.3
Intermediate Shell B4712-1	1.1	89.8	0.434	0.768	0	69.0	0.0	17.0	34.0	103.0
Intermediate Shell B4712-2	1.1	83.2	0.434	0.768	12	63.9	0.0	17.0	34.0	109.9
Using Credible Surveillance Data[(f)]	2.1	102.2	0.434	0.768	12	78.5	0.0	8.5	17.0	107.5[(f)]
Intermediate Shell B4712-3	1.1	73.7	0.434	0.768	10	56.6	0.0	17.0	34.0	100.6
Lower Shell B4713-1	1.1	83.0	0.426	0.763	8	63.3	0.0	17.0	34.0	105.3
Lower Shell B4713-2	1.1	82.4	0.426	0.763	8	62.8	0.0	17.0	34.0	104.8
Lower Shell B4713-3	1.1	82.6	0.426	0.763	10	63.0	0.0	17.0	34.0	107.0
Intermediate to Upper Shell Circumferential Weld Seam 8-442	1.1	205.6	0.00593	0.076	-56	15.6	17.0	7.8	37.4	-3.0
Intermediate to Lower Shell Circumferential Weld Seam 9-442	1.1	91.4	0.419	0.759	-56	69.3	17.0	28.0	65.5	78.9
Upper Shell Longitudinal Weld Seams 1-442 A & C (60° / 300°)	1.1	208.7	0.00387	0.055	-56	11.5	17.0	5.8	35.9	-8.5
Upper Shell Longitudinal Weld Seam 1-442 B (180°)	1.1	208.7	0.00214	0.035	-56	7.2	17.0	3.6	34.8	-14.0
Intermediate Shell Longitudinal Weld Seams 2-442 B & C (120° / 240°)	1.1	189.1	0.311	0.680	-56	128.5	17.0	28.0	65.5	138.1
Intermediate Shell Longitudinal Weld Seam 2-442 A (0°)	1.1	189.1	0.157	0.513	-56	97.0	17.0	28.0	65.5	106.5
Lower Shell Longitudinal Weld Seams 3-442 A & C (60° / 300°)	1.1	208.6	0.307	0.676	-56	141.1	17.0	28.0	65.5	150.6
Using Credible Surveillance Data for Weld Heat # 21935/12008[(f)]	2.1	200.7	0.307	0.676	-56	135.7	17.0	14.0	44.0	123.8[(f)]
Lower Shell Longitudinal Weld Seam 3-442 B (180°)	1.1	208.6	0.156	0.512	-56	106.8	17.0	28.0	65.5	116.3
Using Credible Surveillance Data for Weld Heat # 21935/12008[(f)]	2.1	200.7	0.156	0.512	-56	102.8	17.0	14.0	44.0	90.8[(f)]

Notes for Table 6-3 contained on the next page.

Notes for Table 6-3:

(a) The Regulatory Guide 1.99, Revision 2 [Ref. 4] methodology was utilized in the calculation of the ART values.

(b) CFs are taken from Table 5-3.

(c) Fluence and FFs are taken from Table 6-1.

(d) $RT_{NDT(U)}$ and σ_I values are taken from Table 3-1.

(e) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(f) The credibility evaluation in Appendix A of this report determined that the surveillance data for intermediate shell plate B4712-2 are deemed credible. The surveillance weld data for Heat # 21935/12008 from the Diablo Canyon Unit 2 surveillance program, applicable to the Lower Shell Longitudinal Weld Seams 3-442 A, B, & C are deemed credible in WCAP-17315-NP [Ref. 15]. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF.

Table 6-4 Limiting ART Values for Salem Unit 2 at 50 EFPY

	Limiting ART Values[(a)] (°F)	Limiting Material
1/4T location	191.6[(b)]	Intermediate Shell Longitudinal Weld Seams 2-442 B & C
3/4T location	138.1[(b)]	Intermediate Shell Longitudinal Weld Seams 2-442 B & C

Notes:

(a) Values are the limiting values from Tables 6-2 and 6-3.

(b) Note, the ART value calculated for Lower Shell Longitudinal Weld Seams 3-442 A & C have higher ART values at both the 1/4T and 3/4T locations when a Position 1.1 CF is used. Because credible surveillance data exists from the Diablo Canyon Unit 2 surviellance program, the lower 1/4T and 3/4T ART values, calculated with a Position 2.1 CF, supersede those values.

7 P-T LIMIT CURVE APPLICABILITY EVALUATION

The existing P-T limit curves, implemented in Technical Specifications Figures 3.4-2 and 3.4-3, were calculated in WCAP-15566 [Ref. 19] and are valid through 32 EFPY. In support of license renewal, new P-T limit curves were generated in WCAP-16982-NP [Ref. 1] valid through 50 EFPY. In order to support the implementation of the 50 EFPY P-T limit curves, the applicability of the P-T limit curves is evaluated by comparing the 50 EFPY limiting 1/4T and 3/4T ART values contained in Tables 6-2 and 6-3 with those used in WCAP-16982-NP.

This comparison is provided in Table 7-1.

Table 7-1 Summary of the Limiting ART Values

1/4T Limiting ART (°F)		3/4T Limiting ART (°F)	
WCAP-16982-NP	**Table 6-4**	**WCAP-16982-NP**	**Table 6-4**
209	191.6	150	138.1

Table 7-1 shows that the limiting ART values at the 1/4T and 3/4T locations will not exceed the ART values used in the P-T limit curves of WCAP-16982-NP. Furthermore, the limiting initial RT_{NDT} value of 15°F for the closure head flange and the vessel flange remains appropriate, and the P-T limit curves flange notch and bolt-up temperature require no change or further consideration. Therefore, the EOLE (50 EFPY) P-T limit curves generated in WCAP-16982-NP can be implemented as planned.

The enable temperature presented in WCAP-16982-NP is based on the same ART values as the P-T limit curves. Since the P-T limit curves remain valid, the enable temperature also remains valid. Thus, from WCAP-16982-NP, the minimum required enable temperature without margins for instrument uncertainty is a coolant temperature equal to 274°F through EOLE. With margins for instrument uncertainty (+18°F), the minimum required enable temperature is a coolant temperature equal to 292°F.

<u>Inlet and Outlet Nozzles P-T Limit Curves</u>

NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. 9] requires that the P-T limit curves account for the higher stresses in the nozzle corner region due to the potential for more restrictive P-T limits, even if the RT_{NDT} for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries.

PWROG-15109-NP-A [Ref. 20] addresses this concern generically for the U.S. pressurized water reactor (PWR) operating fleet. The results of PWROG-15109-NP-A demonstrate that P-T limit curves developed with current NRC-approved methods (e.g., WCAP-14040-A [Ref. 21]) bound the generic nozzle P-T limit curves. This document has been approved by the NRC in Refence 22 as an acceptable means to address the concerns of RIS 2014-11. The results and conclusions of PWROG-15109-NP-A are applicable as long as the plant-specific Salem Unit 2 fluence of the nozzle corners remains less than the screening criterion of

4.28×10^{17} n/cm^2 (E > 1.0 MeV), as described in PWROG-15109-NP-A. Section 2 of this report demonstrates Salem Unit 2 adherence to this screening criterion, thus PWROG-15109-NP-A is applicable.

In conclusion, PWROG-15109-NP-A demonstrates that the nozzles will not be limiting with respect to the P-T limit curves at Salem Unit 2. Therefore, the concerns of RIS 2014-11 are adequately addressed.

8 REFERENCES

1. Westinghouse Report WCAP-16982-NP, Revision 3, “Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity,” December 2020.

2. Regulatory Guide 1.190, Revision 0, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” U.S. Nuclear Regulatory Commission, March 2001. *[Agencywide Documents Access and Management System [ADAMS] Accession Number ML010890301]*

3. Westinghouse Report WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018.

4. Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” U.S. Nuclear Regulatory Commission, May 1988. *[ADAMS Accession Number ML003740284]*

5. Radiation Safety Information Computational Center (RSICC) Data Library Collection DLC-185, *BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, Oak Ridge National Laboratory (ORNL), July 1999.

6. ORNL Report ORNL/TM-13205, “Pool Critical Assembly Pressure Vessel Facility Benchmark,” (NUREG/CR-6454), July 1997.

7. ORNL Report ORNL/TM-13204, “H.B. Robinson-2 Pressure Vessel Benchmark,” (NUREG/CR-6453), October 1997 (published February 1998).

8. Code of Federal Regulations 10 CFR 50, Appendix G, “Fracture Toughness Requirements,” U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

9. NRC Regulatory Issue Summary 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” U.S. Nuclear Regulatory Commission, October 2014. *[ADAMS Accession No. ML14149A165]*

10. Combustion Engineering Owners Group Report CE NPSD-1119, Revision 01, “Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Content,” July 1998.

11. Code of Federal Regulations 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

12. Westinghouse Report WCAP-15423, Revision 0, “Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program,” September 2000.

13. Westinghouse Report WCAP-18502-NP, Revision 1, “Salem Unit 1 Heatup and Cooldown Limit Curves for Normal Operation,” July 2020.

14. Electric Power Research Institute (EPRI) 2011 Technical Report, *BWRVIP-173-A: BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials.* Palo Alto, CA: July 2011. 1022835.

15. Westinghouse Report WCAP-17315-NP, Revision 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," July 2011.

16. Westinghouse Report WCAP-15692, Revision 0, "Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," August 2001.

17. K. Wichman, M. Mitchell, and A. Hiser, U.S. NRC Presentation, "Generic Letter 92-01 and RPV Integrity Assessment, Status, Schedule, and Issues," NRC/Industry Workshop on RPV Integrity Issues, February 1998. *[ADAMS Accession Number ML110070570]*

18. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subarticle NB-2300, "Fracture Toughness Requirements for Material."

19. Westinghouse Report WCAP-15566, Revision 1, "Salem Unit 2 Heatup and Cooldown Curves for Normal Operation," February 2001.

20. PWR Owners Group Report PWROG-15109-NP-A, Revision 0, "PWR Pressure Vessel Nozzle Appendix G Evaluation," January 2020. *[ADAMS Accession Number ML20024E573]*

21. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

22. NRC Safety Evaluation "Final Safety Evaluation by the Office of Nuclear Reactor Regulation for Pressurized Water Reactor Owners Group Topical Report PWROG-15109-NP, Revision 0, 'PWR Pressure Vessel Nozzle Appendix G Evaluation' EPID L-2018-TOP-0009," October 31, 2019. *[ADAMS Accession Numbers ML19301D063 & ML19301D160]*

APPENDIX A CREDIBILITY RE-EVALUATION OF THE SALEM UNIT 2 SURVEILLANCE PROGRAM

Regulatory Guide 1.99, Revision 2 [Ref. A-1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Positions 2.1 and 2.2 of Regulatory Guide 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date, there have been four surveillance capsules removed and tested from the Salem Unit 2 reactor vessel. In addition, surveillance data from a sister plant is utilized herein to perform neutron radiation embrittlement calculations.

Table A-1 reviews the five criteria of Regulatory Guide 1.99, Revision 2. The following subsections evaluate each of these five criteria for Salem Unit 2 in order to determine the credibility of the surveillance data for use in neutron radiation embrittlement calculations.

Table A-1 Identification of Credibility Criteria Affected by Fluence Change

Criterion No.	Description
1	Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.
2	Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.
3	When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. A-4].
4	The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.
5	The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR 50, "Fracture Toughness Requirements," [Ref. A-2] as follows:

> *… the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

The Salem Unit 2 reactor vessel consists of the following beltline region materials, which likely would have been considered at the time the surveillance program was designed and licensed:

- Intermediate Shell Plates B4712-1, B4712-2, and B4712-3
- Lower Shell Plates B4713-1, B4713-2, and B4713-3
- Intermediate to Lower Shell Circumferential Weld Seam 9-442
(Heat # 90099, Linde 0091 flux, Lot # 3977)
- Intermediate Shell Longitudinal Weld Seams 2-442 A, 2-442 B, and 2-442 C
(Heat # 13253/20291, Linde 1092 flux, Lot # 3833)
- Lower Shell Longitudinal Weld Seams 3-442 A, 3-442 B, and 3-442 C
(Heat # 21935/12008, Linde 1092 flux, Lot # 3889)

Intermediate Shell Plate B4712-2 has the highest initial RT_{NDT} of the beltline plates and was selected for the surveillance program. The surveillance weld was fabricated with weld wire Heat # 13253 and Linde 1092 flux, Lot # 3833/3774 utilizing the same fabrication practice as that used to fabricate the actual vessel beltline welds. This weld was made of one of the heats and with the same type flux as was used in the intermediate shell longitudinal weld seams.

<u>Thus, the Salem Unit 2 surveillance program meets the intent of this criterion.</u>

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

The surveillance capsule analysis report, WCAP-15692 [Ref. A-5], was reviewed and it was determined that <u>this criterion is met</u>, consistent with the analysis in these documents.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. A-4].

The functional form of the least-squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the plates.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 [Ref. A-3]. At this meeting, the NRC presented five cases. Of the five cases, Case 1 ("Surveillance Data Available from Plant but No Other Source") most closely represents the situation for the Salem Unit 2 surveillance plate and weld materials. Case 5 ("Surveillance Data from Other Sources Only") most closely represents the situation for the Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442 A, 3-442 B, and 3-442 C (Heat # 21935/12008) weld material.

<u>Evaluation of the Salem Unit 2 Data Only (Case 1)</u>

Following the NRC Case 1 guidelines, the Salem Unit 2 surveillance plates and weld metal (Heat # 13253) will be evaluated using the Salem Unit 2 data. Table A-2 provides the calculation of the interim CF for Salem Unit 2. Note that when evaluating the credibility of the surveillance weld data, the measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2 Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values. In addition, only Salem Unit 2 data are being considered; therefore, no temperature adjustment is required.

Table A-2 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using Salem Unit 2 Surveillance Capsule Data Only

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[a] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate B4712-2 (longitudinal)	T	0.273	0.646	61.66	39.84	0.418
	U	0.581	0.848	66.54	56.43	0.719
	X	1.13	1.034	93.82	97.02	1.069
	Y	1.83	1.166	105.69	123.21	1.359
Intermediate Shell Plate B4712-2 (transverse)	T	0.273	0.646	74.83	48.35	0.418
	U	0.581	0.848	98.26	83.33	0.719
	X	1.13	1.034	125.15	129.42	1.069
	Y	1.83	1.166	129.33	150.76	1.359
				SUM:	728.36	7.130
	$CF_{B4712-2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (728.36) \div (7.130) = 102.2°F$					
Salem Unit 2 surveillance weld material (Heat # 13253)	T	0.273	0.646	153.17	98.97	0.418
	U	0.581	0.848	185.94	157.68	0.719
	X	1.13	1.034	195.43	202.10	1.069
	Y	1.83	1.166	200.90	234.20	1.359
				SUM:	692.95	3.565
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (692.95) \div (3.565) = 194.4°F$					

Notes:
(a) Fluence and measured ΔRT_{NDT} taken from Table 4-1.
(b) FF = fluence factor = $f^{(0.28 - 0.10 * \log(f))}$.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Guide 1.99, Revision 2 Position 2.1 is presented in Table A-3.

Table A-3 Surveillance Capsule Data Scatter about the Best-Fit Line Using Only Salem Unit 2 Surveillance Data

Material	Capsule	CF(a) (Slope$_{best-fit}$) (°F)	Capsule Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	FF(c)	Measured ΔRT_{NDT}(b) (°F)	Predicted ΔRT_{NDT}(d) (°F)	Scatter ΔRT_{NDT}(e) (°F)	< 17°F (Base Metal) < 28°F (Weld)
Intermediate Shell Plate B4712-2 (longitudinal)	T	102.2	0.273	0.646	61.66	66.0	4.4	Yes
	U	102.2	0.581	0.848	66.54	86.6	20.1	No
	X	102.2	1.13	1.034	93.82	105.6	11.8	Yes
	Y	102.2	1.83	1.166	105.69	119.1	13.4	Yes
Intermediate Shell Plate B4712-2 (transverse)	T	102.2	0.273	0.646	74.83	66.0	8.8	Yes
	U	102.2	0.581	0.848	98.26	86.6	11.6	Yes
	X	102.2	1.13	1.034	125.15	105.6	19.5	No
	Y	102.2	1.83	1.166	129.33	119.1	10.2	Yes
Salem Unit 2 surveillance weld material (Heat # 13253)	T	194.4	0.273	0.646	153.17	125.6	27.6	Yes
	U	194.4	0.581	0.848	185.94	164.8	21.1	Yes
	X	194.4	1.13	1.034	195.43	201.0	5.6	Yes
	Y	194.4	1.83	1.166	200.90	226.6	25.7	Yes

Notes:
(a) CFs calculated in Table A-2.
(b) Fluence and measured ΔRT_{NDT} values are taken from Table 5-1.
(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(d) Predicted ΔRT_{NDT} = CF x FF.
(e) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

From a statistical point of view, +/- 1σ would be expected to encompass 68% of the data. Table A-3 indicates that six of the eight (75%) surveillance data points fall inside the +/- 1σ of 17°F scatter band for surveillance base metals. Therefore, all the plate data are deemed "credible" per the third criterion.

Table A-3 indicates that all surveillance data points fall inside the +/- 1σ of 28°F scatter band for surveillance weld materials. 100% of the data are bounded; therefore, the surveillance weld data are deemed "credible" per the third criterion. Note that, while credible, the data for surveillance weld Heat # 13253 was determined not to be relevant to the welds in the reactor vessel; therefore, the data are not used for embrittlement calculations of the Salem Unit 2 reactor vessel.

Evaluation of Weld Data from All Sources (Case 5)

Next, data from all sources are considered in order to evaluate the credibility of Heat # 21935/12008 used in the Salem Unit 2 Lower Shell Longitudinal Weld Seams 3-442 A, 3-442 B, and 3-442 C using the NRC Case 5 guidelines.

WCAP-17315-NP [Ref. A-6] already concluded that the data for the Diablo Canyon Unit 2 surveillance weld are credible, therefore, the data are not re-analyzed separately herein.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pad. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Salem Unit 2 surveillance program does not include correlation monitor material. Therefore, this criterion is not applicable to Salem Unit 2.

CONCLUSION:

Based on the preceding responses to the five criteria of Regulatory Gide 1.99, Revision 2 [Ref. A-1], Section B, the Salem Unit 2 surveillance data are credible.

REFERENCES

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A-2 Code of Federal Regulations 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

A-3 K. Wichman, M. Mitchell, and A. Hiser, U.S. NRC Presentation, "Generic Letter 92-01 and RPV Integrity Assessment, Status, Schedule, and Issues," NRC/Industry Workshop on RPV Integrity Issues, February 1998. *[ADAMS Accession Number ML110070570]*

A-4 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

A-5 Westinghouse Report WCAP-15692, Revision 0, "Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," August 2001.

A-6 Westinghouse Report WCAP-17315-NP, Revision 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," July 2011.

APPENDIX B PRESSURIZED THERMAL SHOCK AND EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

B.1 PRESSURIZED THERMAL SHOCK

A limiting condition on RPV integrity known as pressurized thermal shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the RPV under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling on PTS (10 CFR 50.61 [Ref. B-1]) that established screening criteria on pressurized water reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. RT_{PTS} screening values were set by the NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seals for plant operation to the end-of-plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 [Ref. B-2].

These accepted methods were used with the surface fluence values of Section 2 to calculate the following RT_{PTS} values for the Salem Unit 2 RPV materials. The RT_{PTS} calculations are presented in Table B-1 for Salem Unit 2.

PTS Conclusion

All of the reactor vessel materials for Salem Unit 2 are below the RT_{PTS} screening criteria values of 270°F for plates, forgings, and longitudinal welds, and 300°F for circumferentially oriented welds (per 10 CFR 50.61) at 50 EFPY.

Table B-1 RT_{PTS} Calculations for Salem Unit 2 Reactor Vessel Materials at 50 EFPY[(a)]

Material	CF[(b)]	Surface Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surf. FF[(d)]	$RT_{NDT(U)}$[(e)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(f)] (°F)	M (°F)	RT_{PTS} (°F)
Upper Shell B4711-1	73.5	0.0244	0.194	60	14.3	0.0	7.1	14.3	88.6
Upper Shell B4711-2	98.2	0.0244	0.194	60	19.1	0.0	9.5	19.1	98.2
Upper Shell B4711-3	82.6	0.0244	0.194	101	16.0	0.0	8.0	16.0	133.1
Intermediate Shell B4712-1	89.8	2.05	1.196	0	107.4	0.0	17.0	34.0	141.4
Intermediate Shell B4712-2	83.2	2.05	1.196	12	99.5	0.0	17.0	34.0	145.5
*Using **Credible** Surveillance Data*[(g)]	102.2	2.05	1.196	12	122.2	0.0	8.5	17.0	151.2[(g)]
Intermediate Shell B4712-3	73.7	2.05	1.196	10	88.1	0.0	17.0	34.0	132.1
Lower Shell B4713-1	83.0	2.01	1.190	8	98.8	0.0	17.0	34.0	140.8
Lower Shell B4713-2	82.4	2.01	1.190	8	98.1	0.0	17.0	34.0	140.1
Lower Shell B4713-3	82.6	2.01	1.190	10	98.3	0.0	17.0	34.0	142.3
Intermediate to Upper Shell Circumferential Weld Seam 8-442	205.6	0.0280	0.211	-56	43.4	17.0	21.7	55.1	42.5
Intermediate to Lower Shell Circumferential Weld Seam 9-442	91.4	1.98	1.186	-56	108.4	17.0	28.0	65.5	118.0
Upper Shell Longitudinal Weld Seams 1-442 A & C (60°/ 300°)	208.7	0.0183	0.163	-56	34.0	17.0	17.0	48.1	26.0
Upper Shell Longitudinal Weld Seam 1-442 B (180°)	208.7	0.0101	0.110	-56	23.0	17.0	11.5	41.1	8.1
Intermediate Shell Longitudinal Weld Seams 2-442 B & C (120°/ 240°)	189.1	1.47	1.107	-56	209.3	17.0	28.0	65.5	218.8
Intermediate Shell Longitudinal Weld Seam 2-442 A (0°)	189.1	0.741	0.916	-56	173.2	17.0	28.0	65.5	182.7
Lower Shell Longitudinal Weld Seams 3-442 A & C (60°/ 300°)	208.6	1.45	1.103	-56	230.1	17.0	28.0	65.5	239.6
*Using **Credible** Surveillance Data for Weld Heat # 21935/12008*[(g)]	200.7	1.45	1.103	-56	221.4	17.0	14.0	44.0	209.4[(g)]
Lower Shell Longitudinal Weld Seam 3-442 B (180°)	208.6	0.738	0.915	-56	190.8	17.0	28.0	65.5	200.3
*Using **Credible** Surveillance Data for Weld Heat # 21935/12008*[(g)]	200.7	0.738	0.915	-56	183.6	17.0	14.0	44.0	171.6[(g)]

Notes for Table B-1 contained on the next page.

Notes for Table B-1:

(a) The 10 CFR 50.61 [Ref. B-1] methology was utilized in the calculation of the RT_{PTS} values.
(b) CFs are taken from Table 5-3.
(c) Fluence values are taken from Table 2-2.
(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(e) $RT_{NDT(U)}$ and σ_I values are taken from Table 3-1.
(f) Per 10 CFR 50.61, the base metal σ_Δ = 17°F when surveillance data are non-credible or not used to determine the CF, and σ_Δ = 8.5°F when credible surveillance data are used to determine the CF. Also, per 10 CFR 50.61, the weld metal σ_Δ = 28°F when surveillance data are non-credible or not used to determine the CF, and σ_Δ = 14°F when credible surveillance data are used to determine the CF. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.
(g) The credibility evaluation in Appendix A of this report determined that the surveillance data for Intermediate Shell Plate B4712-2 are deemed credible. The surveillance weld data for Heat # 21935/12008 from the Diablo Canyon Unit 2 surveillance program, applicable to the Lower Shell Longitudinal Weld Seams 3-442 A, B, & C, are deemed credible in WCAP-17315-NP [Ref. B-3]. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF.

B.2 EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

The emergency response guideline (ERG) limits, HF04BG [Ref. B-4], were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG – now known as Pressurized Water Reactor Owners Group [PWROG]) to be applicable to all Westinghouse plants.

The highest value of RT_{NDT}, for which the generic category of ERG limits was developed, is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, which is calculated the same way as the RT_{PTS} values are calculated in Section B.1 of this report. The material with the highest RT_{NDT} defines the limiting material. Table B-2 identifies ERG category limits and the limiting material RT_{NDT} values at EOLE for Salem Unit 2.

Table B-2 Evaluation of Salem Unit 2 ERG Limit Category

ERG Pressure-Temperature Limits [Ref. B-4]	
Applicable RT_{NDT} Value[(a)]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category III b
Limiting RT_{NDT} Value[(b)]	
Limiting Reactor Vessel Material	**RT_{NDT} Value @ EOLE**
Salem Unit 2 Intermediate Shell Longitudinal Weld Seams 2-442 B & C	200°F < RT_{NDT} < 250°F

Notes:
(a) Longitudially oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.
(b) Values taken from Table B-1.

Per the ERG limit guidance document [Ref. B-4], some vessels do not change categories for operation through the end of license. However, when a vessel does change ERG categories between the beginning and end of operation, a plant-specific assessment must be performed to determine at what operating time the category changes. Thus, the ERG classification need not be changed until the operating cycle during which the maximum vessel value of actual or estimated real-time RT_{NDT} exceeds the limit on its current ERG category.

Conclusion of ERG P-T Limit Categorization

Per Table B-2, the limiting material for Salem Unit 2 assumes a longitudinally oriented flaw and has an RT_{NDT} value above 200°F but below 250°F at EOLE. Therefore, Salem Unit 2 is limited to ERG Category II through EOLE and will **not** need plant-specific ERG P-T limits.

The Salem Unit 2 RT_{PTS} will exceed 200°F, thus the plant's ERG category changes from Category I to II, when the fluence of the Intermediate Shell Longitudinal Weld Seams 2-442 B & C exceeds 1.03 x 10^{19} n/cm^2 (E > 1.0 MeV). Interpolations of the fluences in Table 2-2 show that this will occur at 34.3 EFPY. Therefore, Salem Unit 2 must switch from ERG Category I to Category II prior to the cycle in which 34.3 EFPY is reached. Salem Unit 2 will then remain ERG Category II through EOLE (50 EFPY).

REFERENCES

B-1 Code of Federal Regulations 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

B-2 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988. *[ADAMS Accession Number ML003740284]*

B-3 Westinghouse Report WCAP-17315-NP, Revision 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations," July 2011.

B-4 Westinghouse Owners Group Report HF04BG, "Background Information for Westinghouse Owners Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 3," March 2014.

APPENDIX C VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

C.1 NEUTRON DOSIMETRY

Comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for all surveillance capsules withdrawn from service to-date are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. C-1]. One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least-squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in this report.

C.2 SENSOR REACTION RATE DETERMINATIONS

In this section, the results of the evaluations of the in-vessel neutron sensor sets withdrawn and analyzed to-date as part of the reactor vessel materials surveillance program are presented.

Eight irradiation capsules attached to the thermal shield were included in the reactor design to constitute the reactor vessel surveillance program. The capsules were located at azimuthal angles of 4° (Capsule S), 176° (Capsule V), 184° (Capsule W), and 356° (Capsule Z) that are 4° from the core cardinal axes and 40° (Capsule T), 140° (Capsule U), 220° (Capsule X), and 320° (Capsule Y) that are 40° from the core cardinal axes. The irradiation history of each of these eight in-vessel surveillance capsules is summarized as follows:

Capsule	Location	Irradiation History
T	40°	Cycle 1 (withdrawn for analysis)
U	40°	Cycles 1-3 (withdrawn for analysis)
X	40°	Cycles 1-6 (withdrawn for analysis)
Y	40°	Cycles 1-11 (withdrawn for analysis)
S	4°	In the reactor
V	4°	In the reactor
W	4°	In the reactor
Z	4°	In the reactor

The azimuthal locations included in the above tabulation represent the first-octant-equivalent (FOE) azimuthal angle of the geometric center of the respective surveillance capsules.

The passive neutron sensors included in the evaluations of the surveillance capsules are summarized as follows:

Sensor Material	Reaction of Interest	Capsule T	Capsule U	Capsule X	Capsule Y
Copper	$^{63}Cu(n,\alpha)\ ^{60}Co$	X	X	X	X
Iron	$^{54}Fe(n,p)\ ^{54}Mn$	X	X	X	X
Nickel	$^{58}Ni(n,p)\ ^{58}Co$	X	X	X	X
Uranium-238	$^{238}U(n,f)\ ^{137}Cs$	X	X	X	X
Neptunium-237	$^{237}Np(n,f)\ ^{137}Cs$	X	X	X	X
Cobalt-Aluminum[(a)]	$^{59}Co(n,\gamma)\ ^{60}Co$	X	X	X[(b)]	X

Notes:
(a) The cobalt-aluminum measurements include both bare wire and cadmium-covered sensors.
(b) The bare cobalt-aluminum measurement for Capsule X was determined to be statistically different than similar measurement data obtained from the 4-loop, thermal-shield reactor plant database for 40° surveillance capsules. As a result, the Capsule X bare cobalt-aluminum measurement was not utilized in the least-squares adjustment for this capsule.

Pertinent physical and nuclear characteristics of the in-vessel surveillance capsule passive neutron sensors are listed in Table C-1.

The use of passive monitors, such as those listed above, does not yield a direct measure of the energy-dependent neutron fluence rate at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron fluence rate has on the target material over the course of the irradiation period. An accurate assessment of the average neutron fluence rate incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor
- The physical characteristics of each monitor
- The operating history of the reactor
- The energy response of each monitor
- The neutron energy spectrum at the monitor location

Results from the radiometric counting of the neutron sensors from the in-vessel capsules are documented in References C-2–C-5, and re-evaluated in this appendix using the RAPTOR-M3G model described in previous sections. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium and neptunium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by the in-vessel capsules was based on monthly power generation data from initial reactor criticality through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The startup and shutdown dates for each cycle of operation used in the evaluations are given in Table C-2.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$R = \frac{A}{N_0 \cdot F \cdot Y \cdot \sum \frac{P_j}{P_{ref}} \cdot C_j \cdot \left(1 - e^{-\lambda \cdot t_j}\right) \cdot e^{-\lambda \cdot t_{d,j}}}$$

where

R	=	Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus)
A	=	Measured specific activity (dps/g)
N_0	=	Number of target element atoms per gram of sensor
F	=	Atom fraction of the target isotope in the target element
Y	=	Number of product atoms produced per reaction
P_j	=	Average core power level during irradiation Period j (MW)
P_{ref}	=	Maximum or reference power level of the reactor (MW)
C_j	=	Calculated ratio of ϕ (E > 1.0 MeV) during irradiation Period j to the time weighted average ϕ (E > 1.0 MeV) over the entire irradiation period
λ	=	Decay constant of the product isotope (1/sec)
t_j	=	Length of irradiation Period j (sec)
$t_{d,j}$	=	Decay time following irradiation Period j (sec)

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variations of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology described in previous sections, accounts for the change in sensor reaction rates caused by variations in fluence rate induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, C_j

is normally taken to be 1.0. However, for multiple-cycle irradiations, particularly those employing low-leakage fuel management, the additional C_j term should be employed. The impact of changing flux levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low-leakage to low-leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel-cycle-specific neutron fluence rate values are used to compute cycle-dependent C_j values at the radial and azimuthal center of the respective capsules at the axial elevation of the active fuel midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, additional corrections were made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the ^{238}U and ^{237}Np sensor reaction rates to account for gamma-ray-induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the fission sensor reaction rates are summarized as follows:

Correction	Capsule T	Capsule U	Capsule X	Capsule Y
^{235}U impurity/Pu build-in	0.8735	0.8616	0.8405	0.8148
^{238}U(γ,f)	0.9555	0.9558	0.9562	0.9564
Net ^{238}U correction	0.8346	0.8235	0.8037	0.7793
^{237}Np(γ,f)	0.9840	0.9840	0.9841	0.9841

These factors were applied in a multiplicative fashion to the decay-corrected uranium and neptunium fission sensor reaction rates.

Additionally, radial gradient correction factors are applied to all samples which are not located at the center of the surveillance capsule. The radial correction factors are computed by calculating the reaction specific responses at the capsule center location and dividing by the corresponding response at the specified radial location. The following table summarizes the radial gradient correction factors.

Reaction	Dosimeter Location	Radius (cm)	Gradient Correction Factor[(a)]			
			Capsule T	Capsule U	Capsule X	Capsule Y
^{63}Cu(n,α) ^{60}Co	Front	211.18	0.960	0.960	0.960	0.959
^{54}Fe(n,p) ^{54}Mn	Middle	211.68	1.054	1.054	1.054	1.055
^{58}Ni(n,p) ^{58}Co	Rear	212.18	1.170	1.171	1.172	1.173
^{238}U(n,f) ^{137}Cs	Center	211.41	--	--	--	--
^{237}Np(n,f) ^{137}Cs	Center	211.41	--	--	--	--
^{59}Co(n,γ) ^{60}Co	Rear	212.18	0.978	0.977	0.977	0.977
^{59}Co(Cd)(n,γ) ^{60}Co	Rear	212.18	1.161	1.160	1.160	1.160

Note:
(a) Gradient correction factors for the ^{238}U and ^{237}Np sensors are not required since they are located at the center of the capsule.

Results of the sensor reaction rate determinations for the in-vessel capsules are given in Table C-3 through Table C-6.

C.3 LEAST-SQUARES EVALUATION OF SENSOR SETS

Least-squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations, resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best estimates for key exposure parameters such as ϕ (E > 1.0 MeV) or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties. For example:

$$R_i \pm \delta_{R_i} = \sum_g \left(\sigma_{ig} \pm \delta_{\sigma_{ig}}\right) \cdot \left(\sigma_g \pm \delta_{\sigma_g}\right)$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the surveillance capsule dosimetry, the FERRET Code [Ref. C-6] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters (ϕ (E > 1.0 MeV) and dpa) along with associated uncertainties for the in-vessel capsules analyzed to-date.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.
3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the plant-specific application of the least-squares methodology, the calculated neutron spectrum was obtained from the results of the neutron transport calculations described in previous sections of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section C.2. The dosimetry reaction cross sections and uncertainties were obtained from the Sandia National Laboratories Radiation Metrology Laboratory (SNLRML) dosimetry cross-section library [Ref. C-7]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018-09, "Application of ASTM Evaluated Cross-Section Data File, Matrix E706 (IIB)" [Ref. C-8].

The uncertainties associated with the measured reaction rates, dosimetry cross sections, and calculated neutron spectrum were input to the least-squares procedure in the form of variances and covariances. The assignment of the input uncertainties followed the guidance provided in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIIA)" [Ref. C-9].

The following provides a summary of the uncertainties associated with the least-squares evaluation of the surveillance capsule sensor sets withdrawn and analyzed to-date.

C.4 REACTION RATE UNCERTAINTIES

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	Uncertainty
$^{63}Cu(n,\alpha)\ ^{60}Co$	5%
$^{54}Fe(n,p)\ ^{54}Mn$	5%
$^{58}Ni(n,p)\ ^{58}Co$	5%
$^{238}U(n,f)\ ^{137}Cs$	10%
$^{237}Np(n,f)\ ^{137}Cs$	10%
$^{59}Co(n,\gamma)\ ^{60}Co$	5%

These uncertainties are given at the 1σ level.

C.5 DOSIMETRY CROSS-SECTION UNCERTAINTIES

The reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections were compiled from the most recent cross-section evaluations, and they have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the plant-specific reactor vessel surveillance program, the following uncertainties in the fission spectrum averaged cross sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
$^{63}Cu(n,\gamma)\ ^{60}Co$	4.08–4.16%
$^{54}Fe(n,p)\ ^{54}Mn$	3.05–3.11%
$^{58}Ni(n,p)\ ^{58}Co$	4.49–4.56%
$^{238}U(n,f)\ ^{137}Cs$ (Cd)	0.54–0.64%
$^{237}Np(n,f)\ ^{137}Cs$ (Cd)	10.32–10.97%
$^{59}Co(n,\gamma)\ ^{60}Co$	0.79–3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

C.6 CALCULATED NEUTRON SPECTRUM

The neutron spectra input to the least-squares adjustment procedure were obtained directly from the results of plant-specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

Using the uncertainties associated with the reaction rates obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g R_{g'} P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and $R_{g'}$ specify additional random group-wise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta e^{-H}$$

where

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g' and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5

Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

C.7 COMPARISONS OF MEASUREMENTS AND CALCULATIONS

This section provides comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations. These comparisons are provided on two levels. In the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. In the second level, calculated values of neutron exposure rates in terms of fast neutron fluence rate ϕ (E > 1.0 MeV) and iron atom displacement rate are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table C-7, comparisons of M/C ratios are listed for the threshold sensors contained in the in-vessel capsules. From Table C-7, it is noted that for the individual threshold sensors, the average M/C ratio ranges from 0.95 to 1.07 with an overall average of 1.01 and an associated standard deviation of 7.0%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table C-8, best-estimate-to-calculation (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate resulting from the least-squares evaluation of each dosimetry set. For the in-vessel capsules, the average BE/C ratio is 0.97 with an associated uncertainty of 4.7% for fast neutron fluence rate (E > 1.0 MeV) and 0.99 with an associated uncertainty of 4.3% for the iron atom displacement rate.

The M/C comparisons based on individual sensor reactions without recourse to the least-squares adjustment procedure are summarized as follows:

Reaction	In-Vessel Capsules	
	Avg. M/C	% Unc. (1σ)
^{63}Cu(n,α) ^{60}Co	1.07	2.8%
^{54}Fe(n,p) ^{54}Mn	0.95	5.8%
^{58}Ni(n,p) ^{58}Co	0.96	5.0%
^{238}U(n,f) ^{137}Cs	0.99	6.7%
^{237}Np(n,f) ^{137}Cs	1.07	4.8%
Linear Average	1.01	7.0%

A similar comparison for exposure rate expressed in terms of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s) are summarized as follows:

Parameter	In-Vessel Capsules	
	Avg. BE/C	**% Unc. (1σ)**
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.97	4.7%
Iron Atom Displacement Rate (dpa/s)	0.99	4.3%

These data comparisons show similar and consistent results, with the linear average M/C ratio of 1.01 in good agreement with the resultant least-squares BE/C ratios of 0.97 for neutron fluence rate (E > 1.0 MeV) and 0.99 for iron atom displacement rate. The comparisons demonstrate that the calculated results provided in previous sections of this report are validated within the context of the assigned 13% uncertainty and, further, show that the ±20% (1σ) agreement between calculation and measurement required by [Ref. C-1] is met.

Table C-1
Nuclear Parameters Used in the Evaluation of the In-Vessel Surveillance Capsule Neutron Sensors

Reaction of Interest	Atomic Weight[a] (g/g-atom)	Target Atom Fraction[b][c]	Product Half-life[b][c][d] (days)	Fission Yield[d] (%)
^{63}Cu(n,α) ^{60}Co	63.546	0.6917	1925.28	N/A
^{54}Fe(n,p) ^{54}Mn	55.845	0.05845	312.13	N/A
^{58}Ni(n,p) ^{58}Co	58.6934	0.68077	70.86	N/A
^{238}U(n,f) ^{137}Cs	238.051	1.00	10975.76	6.02
^{237}Np(n,f) ^{137}Cs	237.048	1.00	10975.76	6.27
^{59}Co(n,γ) ^{60}Co	58.933	0.0015	1925.28	N/A

Notes:

(a) Atomic weight data were taken from the Chart of the Nuclides, 17th Edition, dated 2010 [Ref. C-10].

(b) Half-life and target atom fraction data for ^{63}Cu(n,α), ^{54}Fe(n,p), and ^{58}Ni(n,p), reactions were taken from ASTM Standard E1005-16 [Ref. C-11].

(c) The half-life for the ^{59}Co(n,γ) reaction was taken from ASTM Standard E1005-16 [Ref. C-11]. The target atom fractions for the ^{59}Co(n,γ), ^{238}U(n,f), and ^{237}Np(n,f) reactions are reflective of standard Westinghouse surveillance capsule dosimeter values.

(d) Half-life and fission yield data for the ^{238}U(n,f) and ^{237}Np(n,f) reactions were taken from ASTM Standard E1005-16 [Ref. C-11].

Table C-2
Startup and Shutdown Dates

Cycle	Startup Date	Shutdown Date
1	06/03/1981	01/22/1983
2	07/31/1983	10/04/1984
3	04/13/1985	10/03/1986
4	12/23/1986	08/31/1988
5	11/26/1988	03/31/1990
6	06/24/1990	11/09/1991
7	04/30/1992	03/17/1993
8	06/20/1993	10/13/1994
9	02/04/1995	11/03/1995
10	08/31/1997	04/03/1999
11	05/28/1999	10/05/2000

Table C-3
Measured Sensor Activities and Reaction Rates for Surveillance Capsule T

Monitor Identification	Reaction	Measured Activity[a] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
Cu – Top Mid Wire	^{63}Cu(n,α) ^{60}Co	4.03E+04	3.10E+05	4.73E-17	4.79E-17	4.79E-17
Cu – Mid Wire	^{63}Cu(n,α) ^{60}Co	4.06E+04	3.12E+05	4.76E-17		
Cu – Bot Mid Wire	^{63}Cu(n,α) ^{60}Co	4.16E+04	3.20E+05	4.88E-17		
Fe – Top Wire	^{54}Fe(n,p) ^{54}Mn	7.67E+05	3.22E+06	5.11E-15	5.04E-15	5.04E-15
Fe – Top Mid Wire	^{54}Fe(n,p) ^{54}Mn	7.67E+05	3.22E+06	5.11E-15		
Fe – Mid Wire	^{54}Fe(n,p) ^{54}Mn	7.35E+05	3.09E+06	4.90E-15		
Fe – Bot Mid Wire	^{54}Fe(n,p) ^{54}Mn	7.56E+05	3.17E+06	5.04E-15		
Fe – Bot Wire	^{54}Fe(n,p) ^{54}Mn	7.56E+05	3.17E+06	5.04E-15		
Ni – Top Mid Wire	^{58}Ni(n,p) ^{58}Co	9.30E+05	4.89E+07	7.00E-15	6.99E-15	6.99E-15
Ni – Mid Wire	^{58}Ni(n,p) ^{58}Co	9.23E+05	4.85E+07	6.95E-15		
Ni – Bot Mid Wire	^{58}Ni(n,p) ^{58}Co	9.34E+05	4.91E+07	7.03E-15		
U-238 – Mid	^{238}U(Cd)(n,f) ^{137}Cs	1.23E+05	4.66E+06	3.06E-14	3.06E-14	2.55E-14
Np-237 – Mid	^{237}Np(Cd)(n,f) ^{137}Cs	9.49E+05	3.60E+07	2.26E-13	2.26E-13	2.22E-13
Co-Al – Top	^{59}Co(n,γ) ^{60}Co	7.05E+06	5.52E+07	3.60E-12	3.56E-12	3.56E-12
Co-Al – Bottom	^{59}Co(n,γ) ^{60}Co	6.90E+06	5.40E+07	3.53E-12		
Co-Al – Top (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	2.84E+06	2.64E+07	1.72E-12	1.68E-12	1.68E-12
Co-Al – Bottom (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	2.69E+06	2.50E+07	1.63E-12		

Note:
(a) Measured activities are decay corrected to January 16, 1984.

Table C-4
Measured Sensor Activities and Reaction Rates for Surveillance Capsule U

Monitor Identification	Reaction	Measured Activity[a] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
Cu – Top Mid Wire	$^{63}Cu(n,\alpha)\ ^{60}Co$	7.17E+04	2.85E+05	4.35E-17	4.48E-17	4.48E-17
Cu – Mid Wire	$^{63}Cu(n,\alpha)\ ^{60}Co$	7.40E+04	2.95E+05	4.49E-17		
Cu – Bot Mid Wire	$^{63}Cu(n,\alpha)\ ^{60}Co$	7.59E+04	3.02E+05	4.61E-17		
Fe – Top Wire	$^{54}Fe(n,p)\ ^{54}Mn$	1.00E+06	2.67E+06	4.23E-15	4.37E-15	4.37E-15
Fe – Top Mid Wire	$^{54}Fe(n,p)\ ^{54}Mn$	1.01E+06	2.69E+06	4.27E-15		
Fe – Bot Mid Wire	$^{54}Fe(n,p)\ ^{54}Mn$	1.06E+06	2.83E+06	4.48E-15		
Fe – Bot Wire	$^{54}Fe(n,p)\ ^{54}Mn$	1.06E+06	2.83E+06	4.48E-15		
Ni – Top Mid Wire	$^{58}Ni(n,p)\ ^{58}Co$	5.65E+06	4.05E+07	5.80E-15	5.93E-15	5.93E-15
Ni – Mid Wire	$^{58}Ni(n,p)\ ^{58}Co$	5.76E+06	4.13E+07	5.92E-15		
Ni – Bot Mid Wire	$^{58}Ni(n,p)\ ^{58}Co$	5.90E+06	4.23E+07	6.06E-15		
U-238 – Mid	$^{238}U(Cd)(n,f)\ ^{137}Cs$	2.65E+05	4.57E+06	3.00E-14	3.00E-14	2.47E-14
Np-237 – Mid	$^{237}Np(Cd)(n,f)\ ^{137}Cs$	1.94E+06	3.34E+07	2.10E-13	2.10E-13	2.07E-13
Co-Al – Top	$^{59}Co(n,\gamma)\ ^{60}Co$	1.17E+07	4.74E+07	3.09E-12	3.03E-12	3.03E-12
Co-Al – Bottom	$^{59}Co(n,\gamma)\ ^{60}Co$	1.12E+07	4.54E+07	2.96E-12		
Co-Al – Top (Cd)	$^{59}Co(Cd)(n,\gamma)\ ^{60}Co$	4.57E+06	2.20E+07	1.43E-12	1.42E-12	1.42E-12
Co-Al – Bottom (Cd)	$^{59}Co(Cd)(n,\gamma)\ ^{60}Co$	4.45E+06	2.14E+07	1.40E-12		

Note:
(a) Measured activities are decay corrected to February 2, 1987.

Table C-5
Measured Sensor Activities and Reaction Rates for Surveillance Capsule X

Monitor Identification	Reaction	Measured Activity[(a)] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
Cu – Top Mid Wire	^{63}Cu(n,α) ^{60}Co	1.11E+05	2.49E+05	3.80E-17	3.87E-17	3.87E-17
Cu – Mid Wire	^{63}Cu(n,α) ^{60}Co	1.13E+05	2.54E+05	3.87E-17		
Cu – Bot Mid Wire	^{63}Cu(n,α) ^{60}Co	1.15E+05	2.58E+05	3.94E-17		
Fe – Top Wire	^{54}Fe(n,p) ^{54}Mn	1.24E+06	2.52E+06	4.00E-15	3.77E-15	3.77E-15
Fe – Top Mid Wire	^{54}Fe(n,p) ^{54}Mn	1.15E+06	2.34E+06	3.71E-15		
Fe – Mid Wire	^{54}Fe(n,p) ^{54}Mn	1.14E+06	2.32E+06	3.68E-15		
Fe – Bot Mid Wire	^{54}Fe(n,p) ^{54}Mn	1.18E+06	2.40E+06	3.81E-15		
Fe – Bot Wire	^{54}Fe(n,p) ^{54}Mn	1.13E+06	2.30E+06	3.65E-15		
Ni – Top Mid Wire	^{58}Ni(n,p) ^{58}Co	7.90E+06	3.65E+07	5.23E-15	5.21E-15	5.21E-15
Ni – Mid Wire	^{58}Ni(n,p) ^{58}Co	7.97E+06	3.69E+07	5.28E-15		
Ni – Bot Mid Wire	^{58}Ni(n,p) ^{58}Co	7.75E+06	3.58E+07	5.13E-15		
U-238 – Mid	^{238}U(Cd)(n,f) ^{137}Cs	4.62E+05	3.67E+06	2.41E-14	2.41E-14	1.94E-14
Np-237 – Mid	^{237}Np(Cd)(n,f) ^{137}Cs	3.38E+06	2.68E+07	1.68E-13	1.68E-13	1.66E-13
Co-Al – Top	^{59}Co(n,γ) ^{60}Co	7.13E+06	1.63E+07	1.06E-12	9.80E-13	9.80E-13
Co-Al – Bottom	^{59}Co(n,γ) ^{60}Co	6.03E+06	1.38E+07	8.98E-13		
Co-Al – Bottom (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	6.71E+06	1.82E+07	1.19E-12	1.19E-12	1.19E-12

Note:
(a) Measured activities are decay corrected to March 4, 1992.

Table C-6
Measured Sensor Activities and Reaction Rates for Surveillance Capsule Y

Monitor Identification	Reaction	Measured Activity[a] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
Cu – Top Mid Wire	^{63}Cu(n,α) ^{60}Co	1.09E+05	2.25E+05	3.44E-17	3.51E-17	3.51E-17
Cu – Mid Wire	^{63}Cu(n,α) ^{60}Co	1.11E+05	2.30E+05	3.50E-17		
Cu – Bot Mid Wire	^{63}Cu(n,α) ^{60}Co	1.14E+05	2.36E+05	3.60E-17		
Fe – Top Wire	^{54}Fe(n,p) ^{54}Mn	9.23E+05	2.07E+06	3.28E-15	3.36E-15	3.36E-15
Fe – Top Mid Wire	^{54}Fe(n,p) ^{54}Mn	9.33E+05	2.09E+06	3.32E-15		
Fe – Mid Wire	^{54}Fe(n,p) ^{54}Mn	9.44E+05	2.11E+06	3.35E-15		
Fe – Bot Mid Wire	^{54}Fe(n,p) ^{54}Mn	9.66E+05	2.16E+06	3.43E-15		
Fe – Bot Wire	^{54}Fe(n,p) ^{54}Mn	9.56E+05	2.14E+06	3.40E-15		
Ni – Top Mid Wire	^{58}Ni(n,p) ^{58}Co	4.12E+06	3.40E+07	4.87E-15	4.83E-15	4.83E-15
Ni – Mid Wire	^{58}Ni(n,p) ^{58}Co	4.07E+06	3.36E+07	4.81E-15		
Ni – Bot Mid Wire	^{58}Ni(n,p) ^{58}Co	4.05E+06	3.35E+07	4.79E-15		
U-238 – Mid	^{238}U(Cd)(n,f) ^{137}Cs	6.47E+05	3.30E+06	2.16E-14	2.16E-14	1.69E-14
Np-237 – Mid	^{237}Np(Cd)(n,f) ^{137}Cs	4.71E+06	2.40E+07	1.51E-13	1.51E-13	1.48E-13
Co-Al – Top	^{59}Co(n,γ) ^{60}Co	1.59E+07	3.35E+07	2.19E-12	2.17E-12	2.17E-12
Co-Al – Bottom	^{59}Co(n,γ) ^{60}Co	1.57E+07	3.31E+07	2.16E-12		
Co-Al – Top (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	6.86E+06	1.72E+07	1.12E-12	1.09E-12	1.09E-12
Co-Al – Top (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	6.88E+06	1.72E+07	1.12E-12		
Co-Al – Bottom (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	6.51E+06	1.63E+07	1.06E-12		
Co-Al – Bottom (Cd)	^{59}Co(Cd)(n,γ) ^{60}Co	6.57E+06	1.64E+07	1.07E-12		

Note:
(a) Measured activities are decay corrected to March 26, 2001.

Table C-7
Comparison of Measured and Calculated Threshold Foil Reaction Rates for the In-Vessel Capsules

Reaction	Capsule				Average	Std. Dev.
	T	U	X	Y		
^{63}Cu(n,α) ^{60}Co	1.10	1.08	1.07	1.03	1.07	2.8%
^{54}Fe(n,p) ^{54}Mn	1.03	0.94	0.94	0.90	0.95	5.8%
^{58}Ni(n,p) ^{58}Co	1.03	0.92	0.95	0.94	0.96	5.0%
^{238}U(n,f) ^{137}Cs	1.03	1.06	0.97	0.91	0.99	6.7%
^{237}Np(n,f) ^{137}Cs	1.12	1.11	1.05	1.01	1.07	4.8%
Average of M/C Results					1.01	7.0%

Table C-8
Comparison of Calculated and Best-Estimate Exposure Rates for the In-Vessel Capsules

Capsule	Fast (E > 1.0 MeV) Fluence Rate		Iron Atom Displacement Rate	
	BE/C	Std. Dev.	BE/C	Std. Dev.
T	1.03	6.0%	1.04	7.0%
U	0.98	6.0%	1.00	7.0%
X	0.96	6.0%	0.97	7.0%
Y	0.92	6.0%	0.94	7.0%
Average	0.97	4.7%	0.99	4.3%

C.8 REFERENCES

C-1 Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.

C-2 Westinghouse Report WCAP-10492, Rev. 0, "Analysis of Capsule T from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," March 1984.

C-3 Westinghouse Report WCAP-11554, Rev. 0, "Analysis of Capsule U from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," September 1987.

C-4 Westinghouse Report WCAP-13366, Rev. 0, "Analysis of Capsule X from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," June 1992.

C-5 Westinghouse Report WCAP-15692, Rev. 0, "Analysis of Capsule Y from the Public Service Electric and Gas Company Salem Unit 2 Reactor Vessel Radiation Surveillance Program," August 2001.

C-6 A. Schmittroth, *FERRET Data Analysis Core*, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

C-7 Radiation Safety Information Computational Center (RSICC) Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994.

C-8 ASTM Standard E1018-09, Application of ASTM Evaluated Cross-Section Data File, Matrix E706 (IIB), American Society for Testing and Materials, 2013.

C-9 ASTM Standard E944-13, Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIIA), American Society for Testing and Materials, 2013.

C-10 *Nuclides and Isotopes: Chart of the Nuclides*, 17th Edition, Lockheed Martin, 2010.

C-11 ASTM Standard E1005-16, Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, American Society for Testing and Materials, 2016.

Enclosure 5

Westinghouse Letter Report LTR-SCS-20-52-NP, Rev. 1, “Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis,” (Non-Proprietary)

Westinghouse

To: Tim L. O'Connor

Date: July 26, 2021

From: Functional, Systems & Setpoints Engineering
Phone: 412-374-2779

Our ref: LTR-SCS-20-52-NP, Rev. 1

Subject: **Salem Unit 2 Low Temperature Overpressure Protection System (LTOPS) / Pressurizer Overpressure Protection System (POPS) Analysis**

Reference:

1. LTR-AMER-MKG-18-1118, Revision 1, "Westinghouse Revised Offer for Pressure-Temperature Limits Support and Surveillance Capsule Removal, Testing and Analysis for Salem Units 1 and 2," October 2018.
2. PSE-09-18, Revision 0, "Transmittal of LTOPS Setpoint Evaluation for Salem Units 1 and 2," March 2009.

This letter transmits the Pressurizer Overpressure Protection System (POPS) analysis report for Salem Unit 2 to Public Service Electric and Gas (PSEG) Nuclear. Per the Reference 1 offer, this analysis was performed to validate the applicability of the setpoints in the previous POPS analysis performed in Reference 2. This evaluation confirms that the current POPS pressurizer Power Operated Relief Valve (PORV) settings remain valid. The Westinghouse Non-Proprietary Class 3 version of the analysis is provided in Attachment 1 with the proprietary information identified within brackets removed.

The attachment to this letter will be transmitted to PSEG along with an application for withholding proprietary information from public disclosure and supporting affidavit. The types of proprietary information are identified via superscripts following each bracket, which correspond to the types described in item 5 of the corresponding application for withholding proprietary information.

Please contact the undersigned with any questions.

Luke J. Mitchell*
Functional, Systems & Setpoints Engineering

Verifier: Thomas G. Joseph*
Functional, Systems & Setpoints Engineering

Manager: Steven R. Billman*
Functional, Systems & Setpoints Engineering

Attachment 1: Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis (Non-Proprietary)

*** Electronically approved records are authenticated in the electronic document management system.***

LTR-SCS-20-52-NP, Rev. 1

Attachment 1

Salem Unit 2 Pressurizer Overpressure Protection System (POPS) Analysis

July 2021
(9 Pages)

Luke J. Mitchell
Functional, Systems & Setpoints Engineering

Thomas G. Joseph
Functional, Systems & Setpoints Engineering

1.0 Introduction

The Low Temperature Overpressure Protection System (LTOPS), known as the Pressurizer Overpressure Protection System (POPS) at Salem, provides Reactor Coolant System (RCS) pressure relief capability to mitigate the overpressure transients that may occur during cold shutdown, heatup, and cooldown operations to minimize the potential for challenging reactor vessel integrity limits when operating at low temperature conditions (i.e., 10 CFR 50, Appendix G limits). At Salem, the pressurizer Power Operated Relief Valves (PORVs), with reduced lift settings, provide a method of POP. The POPS PORV setpoints are selected in accordance with the NRC approved methodology in Reference 1 such that the peak pressure during the design basis Mass Injection (MI) and Heat Injection (HI) transients will not exceed the isothermal Appendix G Pressure-Temperature (P-T) limits.

This evaluation will demonstrate that the current POPS analysis documented in Reference 2 remains valid and any changes to the 50 Effective Full-Power Year (EFPY) P-T limit curve applicability term will be reconciled. The current POPS analysis was performed for the 50 EFPY P-T limit curves documented in Reference 3. The 50 EFPY P-T limit curve applicability term was subsequently confirmed to remain valid in Reference 4.

The scope of this evaluation will include reviewing inputs from Reference 1 to determine if any have changed after those analyses were completed. Any changes will be evaluated to determine if POPS PORV settings need to be updated. This work was completed in accordance with the scope defined in Reference 5.

1.1 Limits of Applicability

The results of this evaluation are applicable to Salem Unit 2 with the key input parameters defined in Section 2.1 and the Reference 3 P-T limits, which were shown to remain valid through 50 EFPY in Reference 4.

1.2 Open Items

None.

2.0 Input Parameters and Assumptions

2.1 Key Inputs

The key analysis input parameters used in the development of the Salem Unit 2 POPS setpoints are summarized in the list below. A detailed list of the inputs was sent to PSEG in Reference 6 for confirmation. The confirmation of the key POPS input parameters was received in Reference 7.

1. Design Basis Mass Injection (MI) Transient

The design basis MI event is defined as flow into a water-solid RCS as a result of either, (a) one High Head Safety Injection (HHSI) pump and one Positive Displacement Pump (PDP), or (b) one Intermediate Head Safety Injection (IHSI) pump; whichever is greater. For the analysis performed in Reference 2, a constant IHSI pump flow rate of [ ]^{a,c} was specified as the bounding MI flow rate. PSEG confirmed this value in Reference 7.

2. Design Basis Heat Injection (HI) Transient

The HI transient is defined as the inadvertent startup of one RCP with the SG secondary side a maximum of 50°F hotter than each of the RCS cold leg temperatures. Prior to the RCP start, all loops are inactive and the entire RCS primary side (except for stagnant water in the SG tubes) is assumed to be 50°F cooler than the secondary side. PSEG confirmed this in Reference 7.

3. Pressurizer PORV Stroke and Delay Times

The PORV characteristics from the analysis of record (Reference 2), confirmed in Reference 7, are as follows.

- Valve type = Copes-Vulcan
- Full open C_V (for sub-cooled water discharge) = [ ]^{a,c} gpm / √psi
- Opening Time = [ ]^{a,c} sec (full close to full open)
- Closing Time = [ ]^{a,c} sec (full open to full close)
- Delay Time (open and close) = [ ]^{a,c} sec[1]
- Valve curve (C_V vs. lift) = See Figure 1
- Maximum pressure relief tank backpressure[2] = [ ]^{a,c} psig
- PORV Reset Pressure = FSAR Section 7.6.3.2 implies the valve closes when pressure decreases below the opening pressure (currently 375 psig), [ ]^{a,c}.

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[1] Reference 7 confirmed a signal delay of [ ]^{a,c} seconds. This may not include valve solenoid and seat delays and is bounded by the previously modeled Reference 2 value of [ ]^{a,c} seconds. Therefore, the current delay time of [ ]^{a,c} seconds is maintained for this analysis.

[2] The backpressure is used to calculate the PORV relief capacity as a function of pressurizer pressure. A higher backpressure results in a lower effective PORV capacity and is limiting for the analysis.

a,c

Figure 1: PORV C_V vs. Lift

4. Wide Range Pressure and Temperature Uncertainties

In accordance with the methodology in Reference 1, pressure and temperature uncertainties were applied during the development of the POPS PORV setpoints. The wide range temperature and pressure uncertainties were provided in Reference 7 as follows.

- Pressure uncertainty = [ ]^{a,c} psig
- Temperature uncertainty = [ ]^{a,c} °F

5. Pressure Drop between Reactor Vessel and Wide Range Pressure Transmitter

The following values were confirmed valid for this analysis in Reference 7 since there were no changes in the location of the wide range pressure transmitter and no major changes in the RCS flow rates.

- 4 RCPs running = [ ]^{a,c} psi
- 2 RCPs running = [ ]^{a,c} psi
- 1 RCP running = [ ]^{a,c} psi

6. RCP Operating Restrictions

Per Reference 7, Salem Unit 2 currently utilizes the following restrictions on RCP operation:

- No more than two RCPs may be operating at RCS Temperatures (T_{RCS}) ≤ 200°F
- All four RCPs are allowed to operate at T_{RCS} > 200°F.

7. PORV Piping Limit

In addition to the Appendix G limits, an 800 psig pressure limit is included to address pressurizer PORV piping load conditions associated with subcooled water discharge (confirmed in Reference 7). This limit is recognized as an operational consideration that is accommodated by the POPS in Reference 1. The PORV piping has been generically evaluated for the water hammer loads associated with cyclic water relief at up to 800 psig. Therefore, when the plant is operated water solid, the POPS settings ensure that the pressure does not exceed the design value of 800 psig.

8. Reactor Coolant Pump (RCP) No. 1 Seal ΔP Limits

As discussed in Reference 1, the POPS analysis generally evaluates the minimum RCS pressure following POPS actuation compared to the minimum RCS pressure for RCP operation. This represents an operational consideration intended to maintain the health of the RCP #1 seals. When there is a conflict between meeting the upper limit (i.e., Appendix G P-T limits) and lower limit (i.e., RCP seal limit), the upper limit takes precedence. [

] a,c Therefore, consistent with the current Reference 2 analysis, the RCP seal ΔP limit will not be explicitly evaluated. [

] a,c

9. Current POPS PORV Settings:

The current analysis in Reference 2 developed a single POPS PORV setpoint of 434 psig for 50 EFPY. PSEG confirmed in Reference 7 that Salem Unit 2 will continue to use a single POPS PORV setting value. It should be noted that the current POPS PORV setpoint of 375 psig at Salem Unit 2 remains conservative with respect to the setting developed in Reference 2.

10. Current POPS Enable Temperature:

The current POPS enable temperature documented in the Salem Unit 2 Technical Specifications is 312°F. The minimum EOLE POPS enable temperature defined in Reference 3 is 292°F (includes uncertainty), which was revalidated through 50 EFPY in Reference 4. Therefore, the current enable temperature defined in the Technical Specifications bounds the minimum EOLE POPS enable temperature defined in Reference 4. In addition, the current POPS enable temperature for Salem Unit 1 was calculated to be [ ] a,c °F in Reference 8. If desired for consistency between both units, PSEG can choose to implement the [ ] a,c °F enable temperature for Salem Unit 2 since it bounds both the current enable temperature documented in the Technical Specifications and the minimum EOLE POPS enable temperature defined in Reference 4.

11. Appendix G P-T Limits

The current POPS analysis was performed for the 50 EFPY P-T limit curves documented in Reference 3. The 50 EFPY P-T limit curves developed in Reference 3 were subsequently confirmed to remain valid for implementation in Reference 4.

In accordance with the Reference 1 methodology, the POPS must protect the steady state (isothermal) P-T limits. Therefore, Table 1 summarizes the current isothermal P-T limits used in the calculation of the POPS PORV settings, which remain valid to 50 EFPY.

Table 1: Steady State Appendix G Limits Valid through 50 EFPY for Salem Unit 2 (without Uncertainties)

Salem Unit 2 for 50 EFPY			
RCS Temperature (°F)	**Appendix G limit (psig)**	**RCS Temperature (°F)**	**Appendix G limit (psig)**
60	621	175	799
65	621	180	819
70	621	185	842
75	621	190	866
80	621	195	894
85	621	200	924
90	621	205	957
95	621	210	994
100	621	215	1034
105	621	220	1079
110	621	225	1129
115	621	230	1184
120	621	235	1245
125	621	240	1312
130	621	245	1386
135	621	250	1468
135.1	693	255	1558
140	702	260	1658
145	712	265	1769
150	723	270	1891
155	736	275	2026
160	749	280	2176
165	764	285	2341
170	781		

2.2 Key Assumptions

The following assumptions are applicable for the Salem Unit 2 POPS setpoint analysis. Unless otherwise noted, these assumptions are consistent with the current analysis in Reference 2.

1. It is assumed that the RCS is enclosed by a non-yielding, inelastic boundary. The pressurizer is assumed to be in a water solid condition with the water at the same subcooled temperature as the remainder of the RCS. [ ] a,c
2. Only one PORV is credited to mitigate the low temperature overpressure event to meet the single failure criteria.
3. All MI cases were analyzed at an RCS temperature of 60°F, which is the minimum RCS temperature corresponding to the bolt up temperature in Reference 3. [ ] a,c
4. For the HI transient the entire RCS primary side, with the exception of the water in the SG tubes, is conservatively assumed to be 50°F cooler than the SG secondary side temperature in all four SGs.
5. A single-phase, sub-cooled water discharge through the PORV was assumed.
6. Letdown flow is conservatively assumed to be isolated during the MI and HI transients. [ ] a,c
7. [ ] a,c

3.0 Description of Evaluation

This evaluation is limited in scope, since the purpose is to ensure that no inputs to the Salem Unit 2 POPS analysis have changed since the current analysis in Reference 2 was performed. This evaluation will review the inputs used in the current analysis and ensure that each input is still valid. Therefore, the methodology used for this evaluation will first consist of performing a cursory review of the current analysis in Reference 2. This review will identify all key inputs and how they are applied in developing the current POPS PORV setting. Then, these inputs will be validated against the current design inputs that were confirmed in Reference 7. If the input parameters have changed since the base analyses, the resulting impact will be reviewed to determine if the POPS analysis will need to be revised.

An analysis was performed in Reference 9 to support the Technical Specification LCO 3.4.1.3 and 3.4.1.4 start notes which suggest that an RCP can be started if the pressurizer water volume is less than 1650 ft^3 (approximately 92%) with no limitation on the magnitude of a potential secondary-to-primary temperature asymmetry. The analysis in Reference 9 showed that the analyses performed in Reference 2 with water solid conditions remain limiting for the calculation of the POPS PORV setting and that the LCO 3.4.1.3 and 3.4.1.4 start notes are acceptable.

4.0 Acceptance Criteria

The following acceptance criteria are used to determine the POPS PORV setpoint for Salem Unit 2 per Reference 2, and are reiterated here for this evaluation:

1. The peak RCS pressure resulting from the design basis MI and HI transients shall not exceed the lower of the following for all applicable temperatures:
    - Maximum allowable pressure of the steady-state 10 CFR 50, Appendix G reactor vessel P-T limits
    - 800 psig (PORV inlet pressure discharge piping limit)
2. The minimum RCS pressure resulting from the design basis MI and HI transients should not drop below the RCP No. 1 Seal ΔP limit.

If there is a conflict between satisfying the upper limits (i.e., the minimum of the Appendix G limits and the piping limit) and the lower limits (i.e., the RCP No. 1 Seal ΔP limits), the upper pressure limits will take precedence. As noted in item #8 of Section 2.1, criterion 2 is not explicitly evaluated [ ] a,c.

5.0 Results and Conclusions

Based on the confirmation of key POPS input parameters from PSEG in Reference 7, it is concluded that no input parameters from the current analysis in Reference 2 have been changed. Reference 4 concluded that the 50 EFPY P-T limits in Reference 3 remain valid through 50 EFPY. Since none of the key POPS input parameters have changed, and the NRC approved methodology in Reference 1 also has not changed, it is concluded that the acceptance criteria outlined in Section 4.0 of this evaluation continue to be met for the POPS settings determined in Reference 2. Specifically, a single pressurizer PORV (both PORVs are required to be operable to address a single failure), with the maximum allowable POPS pressurizer PORV setting of ≤ 434 psig, is capable of providing protection across the full temperature range applicable to POP. The setting requires that no more than two Reactor Coolant Pumps (RCPs) be in operation at indicated RCS temperatures ≤ 200°F. This is consistent with current RCP operating restrictions provided in Reference 7.

Since the POPS PORV setting calculated for 50 EFPY remains bounded by the current technical specification setting of 375 psig, PSEG can either choose to maintain the current 375 psig setting or increase the setting to the current maximum allowable setpoint of 434 psig. Increasing the POPS PORV setting would increase operating margin to POPS actuation during heatup and cooldown operations.

The current POPS enable temperature of 312°F documented in the Salem Unit 2 Technical Specifications bounds the minimum EOLE POPS enable temperature of 292°F (includes uncertainty) defined in Reference 3, which was revalidated through 50 EFPY in Reference 4. In addition, the current POPS enable temperature for Salem Unit 1 was calculated to be [ ] a,c°F in Reference 8. The minimum calculated enable temperature for 50 EFPY is less restrictive than both the current Unit 2 value and the revised Unit 1 value. PSEG can therefore implement a POPS enable temperature as low as 292°F, maintain the current technical specification value of 312°F, or use the Unit 1 value of [ ] a,c °F for consistency between units.

6.0 References

1. WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.
2. [ ] a,c
3. WCAP-16982-NP, Revision 2, "Salem Unit 2 Time-Limited Aging Analysis on Reactor Vessel Integrity," July 2020.
4. WCAP-18571-NP, Revision 0, "Verification of the Salem Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," September 2020.
5. [a,c
6.
7.]
8. LTR-SCS-20-28-NP, Revision 1, "Salem Unit 1 Low Temperature Overpressure Protection System (LTOPS) / Pressurizer Overpressure Protection System (POPS) Analysis," February 2021.
9. [ ] a,c

Enclosure 6

Affidavit for Withholding

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Anthony J. Schoedel, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of LTR-SCS-20-52-P, Revision 1 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

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

Executed on: 7/28/2021

Anthony J. Schoedel, Manager
eVinci Licensing & Configuration Management