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

Subject: Response to Request for Additional Information Regarding Scoping of Metal Fatigue for Salem Nuclear Generating Station, Units 1 and 2, dated January 5, 2010

Reference: Letter from Mr. Donnie Ashley (USNRC) to Mr. Thomas Joyce (PSEG Nuclear, LLC) "REQUEST FOR ADDITIONAL INFORMATION REGARDING SCOPING OF METAL FATIGUE FOR THE SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2, AND THE HOPE CREEK GENERATING STATION", dated January 5, 2010

In the referenced letter, the NRC requested additional information related to Section 4.4.3 of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application (LRA). Enclosed are the responses to this request for additional information.

This letter and its enclosure contain no regulatory commitments.

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

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NRR

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

Executed on 2/1/10

Sincerely,

Paul J. Davison

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

Enclosure: Responses to Request for Additional Information

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

Enclosure

Responses to Request for Additional Information related to Section 4.4.3 of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application (LRA)

RAI S 4.4.3-1
RAI S 4.4.3-2
RAI S 4.4.3-3
RAI S 4.4.3-4
RAI S 4.4.3-5
RAI S 4.4.3-6
RAI S 4.4.3-7

RAI S 4.4.3-1

In the Salem Nuclear Generating Station (Salem) LRA, the background information on leak-before-break (LBB) analysis in Section 4.4.3 does not contain enough information to evaluate the application for this analysis.

Please provide the following information:

1. References for the original LBB reports for the LBB-approved piping for both units.

2. Besides the primary loop piping, identify any other piping systems that have been approved for LBB for both units.

PSEG Response:

1. The original LBB analysis for Salem Units 1 and 2 was documented under WCAP-13659, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Salem Generating Station Units 1 and 2," May 1993.

   The Staff approved PSEG's submittal of this analysis under letter dated May 25, 1994, "Leak-Before-Break Evaluation of Primary Loop Piping, Salem Nuclear Generating Station, Units 1 and 2 (TAC Nos. M85799 and M85800)", from James C. Stone, NRC to Steven E. Miltenberger, Public Service Electric & Gas Company (Accession Number: 9406080285). This letter contains the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Eliminating Primary Loop Pipe Rupture as a Design Basis for Salem 1 and 2, Public Service Electric and Gas Company, Salem Nuclear Generating Station, Units 1 and 2, Docket Nos. 50-272 and 50-311.

2. WCAP-13659 (Section 1.1, Purpose) is only applicable to the primary loop piping at Salem Units 1 and 2. LBB has not been approved for any other piping systems at Salem Units 1 and 2.

RAI S 4.4.3-2

On Page 4-49 of the Salem LRA, the applicant discussed 60-year LBB analyses which were based on the steam generator snubber elimination program, steam generator replacement, power uprate, Tavg operating window, and the mechanical stress improvement process application.

Please provide the following information:

1. Reference the report that contains the 60-year analyses.

2. It is not clear whether the 60-year LBB analyses were performed following the same methodology as that of the original LBB analyses or were a study to determine the impact on the LBB piping from various load changes due to changes in operating conditions. Describe in detail the 60-year LBB analyses or submit the analyses for staff review.

PSEG Response:

1. The report that contains the 60-year analysis is WCAP-16958-P, Rev. 0, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Salem Generating Station Units 1 and 2 for the License Renewal Program," March 2009.

2. The 60-year LBB analysis (WCAP-16958-P) is a Salem Units 1 and 2 specific analysis that was performed using the same methodology as that of the original LBB analysis (WCAP-13659). It was not a study to determine the impact on the LBB piping from various load changes due to changes in operating conditions.

   Below is a summary of the LBB analysis for 60 years.

   a. A fracture mechanics evaluation was performed using Salem Units 1 and 2 plant-specific geometry, operating parameters, loadings, and material properties. Inputs from Steam Generator snubber elimination, steam generator replacement, 1.4% power uprate, Tavg operating window, and Mechanical Stress Improvement Process (MSIP) application at the Reactor Vessel inlet and outlet nozzle locations were used in the 60-year LBB analyses for both Salem Units 1 and 2. MSIP has not yet been implemented on the Salem Unit 2 reactor vessel inlet nozzle locations.

   b. Through-wall leakage flaw sizes at the critical locations were determined for a leak rate of 10 gpm, or ten (10) times the leakage detection system capability of 1 gpm for the Salem Units 1 and 2 using the normal loads.

   c. Stability analyses by the LIMIT load method, as discussed in Appendix A in WCAP-13659, were performed at the critical locations using the faulted loads. The stability analyses by J-integral analysis were performed using the faulted loads and considering the effects of thermal aging of the cast stainless steel material. A margin between the leakage flaw size and the critical flaw size of greater than or equal to 2 was demonstrated.

d. The absolute summation method of faulted load combination was applied to the stability analyses. Since crack stability was demonstrated using the absolute summation method of faulted load combination, a margin of 1 on loads was demonstrated.

e. Fatigue crack growth (FCG) analyses for 60-year plant life were performed and the FCG results were shown acceptable. The FCG analyses are based on the Salem Units 1 and 2 generic Nuclear Steam Supply System (NSSS) design transients and cycles, which bound the 60-year cycles for each of the transients used in the 60-year LBB analyses.

In conclusion, it was demonstrated that all of the LBB margins were demonstrated through the period of extended operation. Therefore, the previous LBB conclusions still remain valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the structural design basis of the Salem Units 1 and 2 through the period of extended operation.

RAI S 4.4.3-3

On page 4-50 of the Salem LRA, the applicant stated that the piping systems include cast austenitic stainless steel piping components. Identify each of the cast austenitic stainless steel piping components that are part of the LBB-approved piping.

PSEG Response:

The only cast austenitic stainless steel (CASS) components located within the LBB-approved primary loop piping system are the elbows in the hot leg, cross-over leg, and cold leg for each of the four loops for Salem Units 1 and 2. Refer to Figure 3-2, "Schematic Diagram of Salem Primary Loop Piping Showing Welds Locations", and Section 4.1, "Primary Loop Piping and Fittings Materials" of WCAP-13659.

RAI S 4.4.3-4

Nickel-based Alloy 600/82/182 material in the pressurized water reactor environment has been shown to be susceptible to primary water stress corrosion cracking (PWSCC).

Please provide the following information:

1. Identify any Alloy 82/182 weld metal and Alloy 600 components used in the LBB-approved piping for both units.

2. If LBB piping, identified in the Salem LRA, contains Alloy 600/82/182 material, discuss any measures (such as weld overlays or mechanical stress improvement) that have been or will be implemented to reduce the susceptibility of PWSCC in the LBB piping components.

3. Discuss the inspection history and future inspection frequency of the Alloy 82/182 dissimilar metal butt welds.

PSEG Response:

1. For each of the Salem Units, the four (4) reactor vessel outlet nozzle-to-safe end welds and the four (4) reactor vessel inlet nozzle-to-safe end welds are the only Alloy 82/182 welds located within the LBB-approved piping. The reactor vessel primary inlet nozzle connects the cold leg piping to the reactor vessel and the outlet nozzles connect the reactor vessel to the hot leg piping.

   There are no Alloy 600 components within the LBB-approved piping for Salem Units 1 and 2.

2. During the Fall 2008 outage, Salem Unit 1 implemented Mechanical Stress Improvement Process (MSIP) at the four (4) Alloy 82/182 reactor vessel inlet nozzle-to-safe end welds and the four (4) Alloy 82/182 reactor vessel outlet nozzle-to-safe end welds to mitigate PWSCC.

   During the Fall 2009 outage, Salem Unit 2 implemented MSIP at the four (4) Alloy 82/182 reactor vessel outlet nozzle-to-safe end welds to mitigate PWSCC. The Salem Unit 2 four (4) Alloy 82/182 reactor vessel inlet nozzle-to-safe end welds were not mitigated using MSIP during the Fall 2009 outage. MSIP for these remaining welds is planned for a future Unit 2 refueling outage.

3. The inspection history and future inspection frequency for the Alloy 82/182 dissimilar metal butt welds is provided below.

<u>Inspection History for the Alloy 82/182 Dissimilar Metal Butt Welds</u>

A review of the past examination history for the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds of each of the Salem Units 1 and 2 reactor vessels dating back to the beginning of the 1st Inservice Inspection (ISI) 10-Year interval was performed.

<u>Salem Unit 1</u>

The reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds were examined during the 1st and 2nd ISI 10-Year intervals, included volumetric (Ultrasonic- UT) and surface (Dye Penetrant-PT) examinations. Several of the weld UT examinations documented recordable indications. All of these indications were evaluated against the ASME Section XI IWB-3500 Acceptance Criteria and all welds were found acceptable. No indications were documented during the PT examinations.

During the current 3rd ISI 10-Year interval, the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds were examined by Bare Metal Visual (BMV) examinations during the Fall 2005 refueling outage in accordance with MRP Letter 2004-05, "Needed Action for Visual Inspection of Alloy 82/182 Butt Welds and Good Practice Recommendations for Weld Joint Configurations", 4/2/2004. There was no evidence of leakage identified. The reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds were examined during the Fall 2008 refueling outage by phased array ultrasonic testing (UT) in accordance with the ASME Section XI 1998 edition, 2000 addenda. Of the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds, only one (1) weld had a flaw whose size exceeded the ASME 1998 edition 2000 addenda Code Section XI, IWB-3500 acceptance criteria. This flaw is located in the Reactor Vessel No. 14 outlet nozzle-to-safe end dissimilar metal Alloy 82/182 butt weld. This flaw was determined to be connected to the ID surface. The flaw was evaluated for continued service as required by 1998 edition 2000 addenda Code Section XI, IWB-3600 using the Westinghouse Flaw Evaluation Handbook and found acceptable for continued operation for up to 36 months without the need for repair or mitigation. Since mitigation via MSIP was implemented at the Reactor Pressure Vessel No. 14 outlet nozzle region during the same refueling outage, no repairs will be required. The remaining seven (7) welds had no recordable indications that exceeded the IWB-3500 acceptance criteria.

MSIP was successfully performed on reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds. A post-MSIP phased array ultrasonic testing (UT) in accordance with the ASME Section XI 1998 edition, 2000 addenda was performed with acceptable results.

<u>Salem Unit 2</u>

The reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds were examined during the 1st and 2nd ISI 10-Year intervals, included

Volumetric (Ultrasonic-UT) and Surface (Dye Penetrant-PT) examinations. Several of the weld UT examinations documented recordable indications. All of these indications were evaluated against the ASME Section XI IWB-3500 Acceptance Criteria. All welds were found acceptable. No indications were documented during the PT examinations.

During the current 3rd ISI 10-Year interval, the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds were examined by Bare Metal Visual (BMV) examinations during the Fall 2006 refueling outage in accordance with MRP Letter 2004-05. There was no evidence of leakage identified. The reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds were examined during the Fall 2009 refueling outage by phased array ultrasonic testing (UT) in accordance with the ASME Section XI 1998 edition, 2000 addenda. The reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds had no recordable indications that exceeded the IWB-3500 acceptance criteria.

MSIP was successfully performed only on the reactor pressure vessel outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds. A post-MSIP phased array ultrasonic testing (UT) in accordance with the ASME Section XI 1998 edition, 2000 addenda was performed with acceptable results.

Frequency for Future Inspections for the Alloy 82/182 Dissimilar Metal Butt Welds

Salem Unit 1

For the Salem Unit 1 reactor pressure vessel inlet (4) and outlet (3) nozzle-to- safe-end dissimilar metal Alloy 82/182 butt welds that are Category "C" in accordance with "Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139)", Rev. 1, 50% of these welds will be volumetrically inspected once during the next 6 years. If no cracks are found during these inspections, these welds shall then be inspected according to the approved ISI program schedule consistent with the existing ASME Code examination program or an approved alternative.

For the Salem Unit 1 Reactor Pressure Vessel No. 14 outlet nozzle-to-safe end dissimilar metal Alloy 82/182 butt weld that is Category "G" in accordance with MRP-139-R1 , this weld will be volumetrically inspected twice over the next four (4) refueling outages. If no additional indications or growth are detected after the second examination, the examination schedule continues with the existing Code examination program for unflawed conditions or an approved alternative.

Salem Unit 2

For the Salem Unit 2 reactor pressure vessel inlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds that are Category "E" in accordance with MRP-139-R1, 100% of these welds will be volumetrically inspected every 6 years.

For the Salem Unit 2 reactor pressure vessel outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds that are Category "C" in accordance with MRP-139-R1, 50% of these welds will be volumetrically inspected once during the next 6 years. If no cracks are found during these inspections, these welds shall be then inspected according to the approved ISI program schedule consistent with the existing ASME Code examination program or an approved alternative.

For both Salem Units 1 and 2, future examinations of the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds beyond the above schedule for both Units will be determined by the Salem Nickel Alloy Aging Management Program (Salem LRA Appendix B, Section B.2.2.6). Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 46).

RAI S 4.4.3-5

As part of reviewing the Salem Time-Limited Aging Analysis (TLAA) of the LBB-approved piping, please provide the following information regarding the maintenance of the structural integrity of the LBB piping:

1. Discuss the inspection history and results of the LBB-approved piping.
2. If indications or flaws are remained in service in the LBB piping, discuss how the indications and flaws are monitored to ensure the structural integrity of the pipe to the end of the period of extended operation.
3. Discuss future inspection schedules for each of the LBB pipes, including the inspection of the existing flaws.

PSEG Response:

1. The Salem Units 1 and 2 piping that has been approved for leak-before-break (LBB) for the primary loop (large bore) piping, elbows, and welds comprising of each Units' hot leg, cold leg, and crossover leg are illustrated in Figure 3-2, "Schematic Diagram of Salem Primary Loop Showing Weld Locations" of WCAP-13659.

   The following discussions apply to the Salem Units 1 and 2 LBB-approved piping boundary subject to the Salem Inservice Inspection (ISI) program examinations, including the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds, the primary loop piping stainless steel welds, and the cast austenitic stainless steel (CASS) elbows.

   A review of the past inspection history, dating back to the beginning of the 1st ISI 10-Year interval for Salem Units 1 and 2 indicates that the welds were examined using surface (Dye Penetrant-PT) and volumetric (Ultrasonic-UT) methods. A review of the surface examination (PT) results found two welds with surface indications that required corrective action. Both of these welds had the indications removed by light surface buffing, and re-examination (PT) found both welds acceptable. These surface indications were not characterized as service-induced flaws.

A review of the volumetric (UT) examination results found some welds with recordable indications. Except for one (1) weld in the Salem Unit 1 Reactor Pressure Vessel No. 14 outlet nozzle-to-safe end dissimilar metal Alloy 82/182 butt weld, these volumetric recordable weld indications were determined to be either geometric indications or acceptable weld flaws that did not exceed the acceptance criteria of ASME Section XI IWB-3500. There were no weld indications found that required corrective action (i.e., repair or replacement). The Salem Unit 1 Reactor Pressure Vessel No. 14 outlet nozzle-to-safe end dissimilar metal Alloy 82/182 butt weld indication that exceeded the acceptance criteria of ASME Section XI IWB-3500 was further evaluated and found acceptable for continued service. The flaw was characterized Primary Water Stress Corrosion Crack (PWSCC) service induced flaw that was connected to the ID surface. This weld was further evaluated for application of MSIP and found acceptable to perform MSIP. Post-MSIP volumetric inspection revealed that the previously rejectable flaw had been reduced to an acceptable level indication per the ASME Section XI IWB-3500 acceptance criteria.

2. All selected ASME Section XI LBB welds are inspected on a periodic basis in accordance with the ASME Section XI, 1998 edition, including the 2000 Addenda, and the approved Risk Informed In-service Inspection Program (RI-ISI). Those welds within the LBB scope that contain Alloy 82/182 weld material are also examined in accordance with the requirements of MRP-139-R1. These selected weld examinations will continue through the end of the current 3rd ISI 10-Year intervals for both Salem Units 1 and 2.

   The Reactor Pressure Vessel No. 14 outlet nozzle-to-safe end dissimilar metal Alloy 82/182 butt weld that contained PWSCC cracking will be re-examined in the Fall 2011 and Fall 2014 refueling outages in accordance with MRP-139-R1 requirements. If the examinations show no indication of crack growth or new cracking, the weld will be placed back into the Salem Risk Informed In-service Inspection Program (RI-ISI) program for future inspections.

3. Both Salem Units are currently in their 3rd ISI 10-Year interval. The Salem Units 1 and 2 primary loop piping that has been approved for LBB is currently subject to inspection in accordance with ASME Section XI, 1998 edition, including the 2000 Addenda, and the approved Risk Informed In-service Inspection Program (RI-ISI) as well as the requirements in MRP-139-R1. These scheduled examinations will continue until the end of the current 3rd ISI 10-Year interval. Following completion of the current 3rd ISI 10-Year interval, the Salem Units 1 and 2 ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD Program will be updated as required by 10 CFR 50.55a, and the examinations will be conducted accordingly. The weld inspection requirements contained in MRP-139-R1 will continue into the next (4th) ISI 10-Year interval for both Salem Unit 1 and 2 until all requirements have been satisfied, and then the welds are placed back into the approved ISI/RI-ISI program for future inspections.

   For both Salem Units 1 and 2, future examinations of the reactor pressure vessel inlet (4) and outlet (4) nozzle-to-safe-end dissimilar metal Alloy 82/182 butt welds beyond the above schedule for both Units will be determined by the Salem Nickel Alloy Aging Management Program (Salem LRA Appendix B, Section B.2.2.6). Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 46).

   For both Salem Units 1 and 2, future examinations of the primary loop piping stainless steel welds beyond the above schedule for both Units will be determined by the Salem ASME

Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Aging Management Program (Salem LRA Appendix B, Section B.2.1.1). Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 1).

In addition, aging of the CASS elbows will be managed with the new Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Aging Management Program (Salem LRA Appendix B, Section B.2.1.6). Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 6).

RAI S 4.4.3-6

On page 4-50 of the Salem LRA, the applicant stated that the numbers of design cycles assumed in the LBB analyses bound the numbers of design cycles projected for 60 years of operation. Discuss how the design cycles assumed in the LBB analysis are verified to ensure that they bound the numbers of design cycles projected for 60 years of operation.

PSEG Response:

The process for determining the 60-year cycles projections is described in LRA Section 4.3.1, subsection "60-Year Transient Projection Methodology", beginning at the bottom of LRA page 4-25. The results of the 60-year projections for each design transient are provided in the third column, "60-Year Projected Cycles", in Salem LRA Tables 4.3.1-3 and 4.3.1-4 for Salem Units 1 and 2, respectively. The 60-year LBB analyses use the Nuclear Steam Supply System (NSSS) design cycle limit for the transients that are listed in the fourth column, "NSSS Design Limit", of Salem LRA Tables 4.3.1-3 and 4.3.1-4 for Salem Units 1 and 2, respectively. Salem LRA Tables 4.3.1-3 and 4.3.1-4 compare the 60-year projected cycles to their respective NSSS design cycle limit for Salem Units 1 and 2, respectively, and confirm that the NSSS design cycle limit bounds the corresponding 60-Year projected cycles for the transients used in the 60-year LBB analyses.

RAI S 4.4.3-7

On page 4-50 of the Salem LRA, under Disposition: Validation, the applicant stated that "...The [LBB] analyses remain valid for the period of extended operation..." Discuss how the LBB analyses are verified to demonstrate that they remain valid for the period of extended operation.

PSEG Response:

Salem Units 1 and 2 will implement a Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program (Salem LRA Appendix B, Section B.3.1.1), which continues to count cycles for each of the transients. An annual report summarizes the current cycles and compares the cumulative values to the design limits. Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 47).

To verify that the number of design cycles for each transient assumed in the LBB analysis continue to bound the number of cycles projected for 60 years of operation, Salem Units 1 and 2 use a cycle counting procedure that monitors each of the design transients. On an annual basis, a report is generated via the procedure that compares the accumulated cycle count to a reportable value of cycles, which is less than the design limit for cycles, for each of the design transients.