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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 NRC Request for Additional Information, dated November 22, 2010, related to 1) The use of the WESTEMS™ Program in Metal Fatigue Analysis, and 2) Confirmation of Environmental Fatigue Locations, associated with the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application

Reference: Letter from Ms. Bennett Brady (USNRC) to Mr. Thomas Joyce (PSEG Nuclear, LLC) "REQUEST FOR ADDITIONAL INFORMATION FOR SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION FOR USE OF WESTEMS PROGRAM IN METAL FATIGUE ANALYSIS (TAC NO. ME1834 AND ME1836)", dated November 22, 2010

In the referenced letter, the NRC Staff requested additional information related to the use of WESTEMS™ at Salem. Additionally, the Staff requested information to confirm that the locations selected for environmental fatigue analysis are the most limiting and bounding for the plant. Enclosure A contains the response to this request for additional information.

Enclosure B provides an update to the License Renewal Commitment List (LRA Appendix A, Section A.5) adding commitment #52 as a result of this RAI response.

As described in the Enclosure, PSEG Nuclear, LLC is currently performing a benchmarking evaluation for both the Unit 2 Pressurizer Surge Nozzle and 1.5-inch Boron Injection Tank (BIT) Safety Injection Nozzle as also requested in the referenced letter. As agreed with the NRC License Renewal Project Manager, the results of this benchmarking effort will be provided to the NRC in a separate submittal by January 7, 2011.

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 12/21/10

Sincerely,

Paul J. Davison

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

Enclosures: A. Response to Request for Additional Information
B. Update to License Renewal Commitment List

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

Enclosure A

Response to Request for Additional Information Related Metal Fatigue associated with the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application

RAI 4.3-07
RAI 4.3-08

RAI 4.3-07:

Background:

Section 4.3.1 of the Salem Nuclear Generating Station, Units 1 and 2 (Salem), license renewal application (LRA) mentions that data from the WESTEMS® fatigue monitoring software were reviewed with respect to pressurizer heatups and cooldowns. Section 4.3.4.2 of the Salem LRA credits the WESTEMS® code for evaluation of fatigue for the pressurizer and surge line locations. Sections A.3.1.1 and B.3.1.1 of the Salem LRA identify that WESTEMS® computes cumulative usage factors for select locations under a discussion of the Metal Fatigue of Reactor Pressure Boundary Program. Section A.4.3.4.2 of the Salem LRA mentions that WESTEMS® was used to evaluate pressurizer insurge/outsurge transients and surge line stratification on the pressurizer.

Issue:

The staff is not clear on the specific use of WESTEMS® at Salem. In addition, the staff has identified concerns regarding the results determined by the WESTEMS® program as a part of the ASME Code fatigue evaluation process. For example, Westinghouse's response to NRC questions regarding the AP1000 Technical Report (see ADAMS Accession No. ML102300072, dated August 13, 2010), describes the ability of users to modify intermediate data (peak and valley stresses/times) used in the analyses. In addition, a response provided on August 20, 2010, (ADAMS Accession No. ML102350440) describes different approaches for summation of moment stress terms. These items can have significant impacts on calculated fatigue cumulative usage factor (CUF). The potential impact for modifications such as these formed the basis for the staff's conclusions in Regulatory Issue Summary (RIS) 2008-30, "Fatigue Analysis of Nuclear Power Plant Components," dated December 16, 2008, where it was noted that simplification of the analysis requires a great deal of judgment by the analyst to ensure that the simplification still provides a conservative result. The staff recognizes that WESTEMS® has been developed under a formal Quality Assurance Program with supporting Technical Bases; however, it is difficult to ascertain the accuracy or conservatism of a location-specific application of WESTEMS® given that a variety of analyst judgments may still be applied to the software outputs by the user on a case-specific basis.

Request:

The staff requests that the licensee provide clarification on the use of WESTEMS® at Salem, as follows:

- Please clarify how WESTEMS® is used at each Salem unit, especially with regard to the Metal Fatigue of Reactor Pressure Boundary Program. Specifically, what transients and locations are monitored by WESTEMS®, what WESTEMS® stress modules are used, and are the stress models used at each Salem unit identical?
- Please describe whether the issues raised in ADAMS Accession Nos. ML102300072 dated August 13, 2010, and ML102350440 dated August 20,

2010, are applicable to each Salem WESTEMS® monitored location. If not, please describe the reasons those issues are not applicable.

- For each location monitored by WESTEMS®, please describe the historical fatigue analyses of record starting from the original ASME Code, Section III design basis fatigue analysis of record. For each follow-on analysis, please describe the reason for the re-analysis, whether the evaluation was referenced in the current licensing basis (CLB), and whether an updated ASME Code, Section III Design Specification and Code Reconciliation were performed in accordance with ASME Code, Section III requirements. Please describe how these analyses are reflected in the results tabulated in Tables 4.3.1-1, 4.3.4-1, 4.3.7-1 and 4.3.7-2.
- Please describe the environmentally-assisted fatigue (EAF) analyses performed for each monitored location, if any.
- Please describe the differences between the stress models used in WESTEMS® and the stress models used in the currently governing fatigue analysis of record and the EAF analysis of record (if any) for each monitored location.
- Please describe how the transient counting results tabulated in Tables 4.3.1-3 and 4.3.1-4 are incorporated into the fatigue results shown in Tables 4.3.7-1 and 4.3.7-2.

In addition, the staff requests benchmarking evaluations for two of the limiting locations monitored in the Salem WESTEMS® application using the same input parameters and assumptions as those used in traditional, ASME Code, Section III CUF calculations for each location. If such calculations do not exist for either of the selected locations, they should be developed using techniques that allow independent comparison with the WESTEMS® results. The intent of this benchmarking evaluation is to confirm that the results of the WESTEMS® models, including any analyst judgments, are acceptable and comparable to traditional ASME Code, Section III analyses for the selected monitored locations.

For the pressurizer surge nozzle and the 1.5" BIT line locations that Salem has indicated are monitored in WESTEMS®, provide a summary of the benchmarking evaluation that includes the following information:

- A comparison of the calculated stresses and CUF using WESTEMS® to the same results from the ASME Code, Section III CUF calculations for all transient pairs representing at least 75 percent of the total CUF from the ASME Code, Section III CUF calculations. One comparison for each unique stress model used in WESTEMS® for each selected location is sufficient.
- Describe the differences in the results between the WESTEMS® evaluation and the ASME Code, Section III CUF calculations for each selected location, and provide a justification for acceptability of the differences.

Salem Response:

Due to the complexity of the request, Salem Nuclear Generating Station Units 1 and 2 (Salem) will respond to each of the bullets requested in numerical order as follows.

Bullet # 1 – Clarification on the use of WESTEMS™ at each Salem Unit

1. To support the Salem License Renewal Application (LRA), Westinghouse had used WESTEMS™ to prepare the environmentally-assisted fatigue (EAF) calculations for the following NUREG/CR-6260 locations for an older vintage Westinghouse plant; (1) Pressurizer Surge Line Nozzle, (2) Surge Line Hot Leg Nozzle, (3) 3-inch Charging Nozzles, (4) 1.5-inch Boron Injection Tank (BIT) Safety Injection Nozzles, and (5) 10-inch Accumulator/Residual Heat Removal (RHR) Cold Leg Nozzles. NUREG/CR-6260 is titled, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components."

   In addition to these calculations, the Salem Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Salem LRA, Appendix B, Section 3.1.1) will use WESTEMS™ as an online tool using input from the Plant Information (PI) computer system (i.e., flows, temperatures, and pressures, etc.). Online data will be monitored on the Chemical and Volume Control (CVC) System, Residual Heat Removal (RHR) System, Safety Injection (SI) System, accumulator injection lines, and the pressurizer surge line to establish thermal hydraulic and mechanical simulation of these systems to create synthetic points representing calculated pressures, flows, temperatures, and moments. These synthetic points will then be used by WESTEMS™ to calculate stresses at specific locations for Salem Units 1 and 2. WESTEMS™ analyzes the stress time histories of the various system locations to evaluate the fatigue damage effects according to the methods defined in ASME Section III, Division I NB-3200 Code criteria (NB-3200).

   As discussed in the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program, additionally, Salem will also use manual cycle counting to monitor design basis transients for other Class 1 piping and components not monitored by WESTEMS™. Although WESTEMS™ has the capability, Salem does not currently use the software to monitor (i.e., count) transients, but only to monitor the parameters that change as a result of a transient to compute stresses and fatigue at the monitored locations.

For each Salem Unit, the following six (6) locations will be monitored by WESTEMS™ to compute fatigue usage:

(1) Pressurizer Surge Line Nozzle Safe End to Pipe Weld (one location per Unit)

(2) Surge Line Hot Leg Nozzle to Pipe Weld (one location per Unit)

(3) Residual Heat Removal (RHR)/ Accumulator Nozzle to Pipe Weld (four separate locations per Unit)

(4) Normal and Alternate Charging Line Nozzles to Pipe Weld (two locations per Unit, one for each nozzle)

(5) Safety Injection Boron Injection Tank (BIT) Nozzle to Pipe Weld (four separate locations per Unit)

(6) Unit 1 Auxiliary Feedwater Nozzle Transition Piece (four separate locations)

    As discussed in the Salem response to RAI 4.3-04, PSEG letter LR-N10-0243, dated July 13, 2010, this location will be modeled into WESTEMS™ to monitor fatigue usage.

For the first five (5) monitoring locations listed above, WESTEMS™ uses the ASME Section III NB-3200 stress module. For the WESTEMS™ online monitoring of the Unit 1 Auxiliary Feedwater Nozzle Transition Piece, Salem will use a monitoring model consistent with the stress model employed in the governing fatigue analysis of record.

For the Residual Heat Removal (RHR)/ Accumulator Nozzle to Pipe Weld, Normal and Alternate Charging Line Nozzles to Pipe Weld, and the Safety Injection Boron Injection Tank (BIT) Nozzle to Pipe Weld locations, the stress models for both units are identical.

There is a slight difference in the respective Unit's stress models for the Pressurizer Surge Line Nozzle Safe End to Pipe Weld location since the Unit 1 surge line is 14-inch schedule 140 piping and has a SA-182 F316 safe end, while the Unit 2 surge line is 14-inch schedule 160 piping and has a SA-182 F316L safe end. For the Surge Line Hot Leg Nozzle to Pipe Weld location, there is a small difference in the stress models also due to the difference in the hot leg nozzle geometry at the surge line connection since there is a difference in piping schedules between the Units 1 and 2 surge lines.

The Auxiliary Feedwater Nozzle Transition Piece is only applicable to Salem Unit 1 since this component does not exist in the Unit 2 auxiliary feedwater nozzle design.

<u>Bullet # 2 – Open Items on the use of WESTEMS™ in Westinghouse AP1000 Design Report</u>

The issues raised in ADAMS Accession Nos. ML102300072 dated August 13, 2010, and ML102350440 dated August 20, 2010, are not applicable to any Salem WESTEMS™ monitored location.

ADAMS Accession No. ML102300072 dated August 13, 2010 has two open items; OI-SRP3.9.1-EMB-05 R3 and OI-SRP3.9.1-EMB-06 R2. Both of these items pertain to the WESTEMS™ NB-3600 module. The WESTEMS™ tool that was used for the Salem EAF calculations, and will be used to monitor online fatigue usage at Salem utilizes the NB-3200 module, and the concerns discussed in the two open items are not applicable to the WESTEMS™ NB-3200 module.

ADAMS Accession No. ML102350440 dated August 20, 2010 has one open item; OI-SRP3.9.1-EMB1-07 R3. This item pertains to the ability of the user to modify the stress peak and valley times selected for inclusion in the fatigue calculations during the process of design fatigue evaluations. The Salem WESTEMS™ online fatigue monitoring tool does not have user capability to modify the stress peak and valley times used in the online fatigue calculations, and therefore, the issues in this letter do not apply to the Salem online use of WESTEMS™. The Salem EAF calculations were performed by Westinghouse using WESTEMS™. Their fatigue calculations did involve limited adjustment to the stress peak and valley times, specifically, redundant stress peaks were removed from the fatigue analyses that resulted from the same transient excursions. The removal of these redundant stress peaks were technically justified and verified, and documented in the calculations, and were considered to have an insignificant impact on the final cumulative fatigue usage, and would not result in any CUF exceeding 1.00.

<u>Bullet # 3 – Historical Fatigue Analyses of Record</u>

Below is a summary table on the history of fatigue analyses prepared for each of the locations monitored by WESTEMS™. Note that the pre-1976 date refers to original design of each of the Salem Units.

WESTEMS™ Monitoring "Limiting" Location	Historical Fatigue Analyses	LRA Table Input/Results
Pressurizer Surge Line Nozzle Safe End to Pipe Weld	Pre-1976 – Original Design considered ASA Piping Code B31.1, 1955 for Unit 1 and USAS Piping Code B31.1, 1967 for Unit 2 (no explicit fatigue analysis). 1992 – WCAP-12914 (Units 1&2) evaluated the Pressurizer Surge Nozzle Safe End to Pipe Weld using ASME Section III Code, 1986 edition. The reason for this re-analysis was in response to NRC Bulletin 88-11. This analysis is referenced in the current licensing basis (CLB). 2003 – WCAP-16194 (Units 1&2) evaluated the entire nozzle assembly for the effects of pressurizer insurge/outsurge transients as recommended by the Westinghouse Owners Group, and not as a result of a regulatory commitment, using ASME Section III Code, 1989 edition. This analysis is not referenced in the CLB. 2009 – WCAP-16994-P and WCAP-16995-P evaluated the entire nozzle assembly using ASME Section III Code, 1986 edition, and determined the safe end to pipe weld as the limiting location for analysis and monitoring for Salem Units 1 and 2, respectively. These WCAPs also include the EAF, and are considered the governing fatigue analyses. The reason for this re-analysis was in support of the Salem LRA. These analyses are not referenced in the CLB.	The WCAP-16194 report also evaluated the Salem Units 1 and 2 Pressurizer Lower Head and Surge Nozzle, where the Code Edition of 1989 is reflected in LRA Table 4.3.1-1. The WCAP-12914 report evaluated the Salem Units 1 and 2 Pressurizer Surge Nozzle Safe End to Pipe Weld location, where the Code Edition of 1986 is reflected in LRA Table 4.3.4-1. The projected 60-year cumulative usage factor (CUF), fatigue life correction factor (Fen), and overall environmentally-assisted CUF are listed in Tables 4.3.7-1 and 4.3.7-2 for Salem Units 1 and 2, respectively. The input to these tables is from WCAP-16994-P and WCAP-16995-P, with a Code Edition of 1986.

WESTEMS™ Monitoring "Limiting" Location	Historical Fatigue Analyses	LRA Table Input/Results
Surge Line Hot Leg Nozzle to Pipe Weld	Pre-1976 – Original Design considered ASA Piping Code B31.1, 1955 for Unit 1 and USAS Piping Code B31.1, 1967 for Unit 2 (no explicit fatigue analysis). 1992 – WCAP-12914 (Units 1&2) evaluated the Surge Line Hot Leg Nozzle to Pipe Weld using ASME Section III Code, 1986 edition. The reason for this re-analysis was in response to NRC Bulletin 88-11. This analysis is referenced in the CLB. 2009 – WCAP-16994-P and WCAP-16995-P evaluated the entire nozzle assembly using ASME Section III Code, 1986 edition, and determined the safe end to pipe weld as the limiting location for analysis and monitoring for Salem Units 1 and 2, respectively. These WCAPs also include the EAF, and are considered the governing fatigue analyses. The reason for this re-analysis was in support of the Salem LRA. These analyses are not referenced in the CLB.	None of the fatigue analyses described are reflected in LRA Table 4.3.1-1. The WCAP-12914 report evaluated the Salem Units 1 and 2 Pressurizer Hot Leg Nozzle to Pipe Weld location, where the Code Edition of 1986 is reflected in LRA Table 4.3.4-1. The projected 60-year cumulative usage factor (CUF), fatigue life correction factor (Fen), and overall environmentally-assisted CUF are listed in Tables 4.3.7-1 and 4.3.7-2 for Salem Units 1 and 2, respectively. Input for these tables is from WCAP-16994-P and WCAP-16995-P with a Code Edition of 1986.

WESTEMS™ Monitoring "Limiting" Location	Historical Fatigue Analyses	LRA Table Input/Results
Residual Heat Removal (RHR)/ Accumulator Nozzle to Pipe Weld	Pre-1976 – Original Design considered ASA Piping Code B31.1, 1955 for Unit 1 and USAS Piping Code B31.1, 1967 for Unit 2 (no explicit fatigue analysis). 2009 – WCAP-16994-P and WCAP-16995-P evaluated the entire nozzle assembly using ASME Section III Code, 1986 edition, and determined the safe end to pipe weld as the limiting location for analysis and monitoring for Salem Units 1 and 2, respectively. These WCAPs also include the EAF, and are considered the governing fatigue analyses. The reason for this re-analysis was in support of the Salem LRA. These analyses are not referenced in the CLB.	None of the fatigue analyses described are reflected in LRA Table 4.3.1-1. The fatigue analyses for the RHR Accumulator Nozzle to Pipe Weld location contained in the WCAP-16994-P and WCAP-16995-P reports are not reflected in LRA Table 4.3.4-1. The projected 60-year cumulative usage factor (CUF), fatigue life correction factor (Fen), and overall environmentally-assisted CUF are listed in Tables 4.3.7-1 and 4.3.7-2 for Salem Units 1 and 2, respectively. Input for these tables is from WCAPs-16994-P and WCAP-16995-P with a Code Edition of 1986.

WESTEMS™ Monitoring "Limiting" Location	Historical Fatigue Analyses	LRA Table Input/Results
Normal and Alternate Charging Nozzle to Pipe Weld	Pre-1976 – Original Design considered ASA Piping Code B31.1, 1955 for Unit 1 and USAS Piping Code B31.1, 1967 for Unit 2 (no explicit fatigue analysis). 2009 – WCAP-16994-P and WCAP-16995-P evaluated the entire nozzle assembly using ASME Section III Code, 1986 edition, and determined the safe end to pipe weld as the limiting location for analysis and monitoring for Salem Units 1 and 2, respectively. These WCAPs also include the EAF, and are considered the governing fatigue analyses. The reason for this re-analysis was in support of the Salem LRA. These analyses are not referenced in the CLB.	None of the fatigue analyses described are inputs to LRA Table 4.3.1-1. The fatigue analyses for the Normal and Alternate Charging Nozzle to Pipe Weld location contained in the WCAP-16994-P and WCAP-16995-P reports are not reflected in LRA Table 4.3.4-1. The projected 60-year cumulative usage factor (CUF), fatigue life correction factor (Fen), and overall environmentally-assisted CUF are listed in Tables 4.3.7-1 and 4.3.7-2 for Salem Units 1 and 2, respectively. Input for these tables is from WCAP-16994-P and WCAP-16995-P with a Code Edition of 1986.

WESTEMS™ Monitoring "Limiting" Location	Historical Fatigue Analyses	LRA Table Input/Results
Safety Injection Boron Injection Tank (BIT) Nozzle to Pipe Weld	Pre-1976 – Original Design considered ASA Piping Code B31.1, 1955 for Unit 1 and USAS Piping Code B31.1, 1967 for Unit 2 (no explicit fatigue analysis). 2009 – WCAP-16994-P and WCAP-16995-P evaluated the entire nozzle assembly using ASME Section III Code, 1986 edition, and determined the nozzle to pipe weld as the limiting location for analysis and monitoring for Salem Units 1 and 2, respectively. These WCAPs also include the EAF, and are considered the governing fatigue analyses. The reason for this re-analysis was in support of the Salem LRA. These analyses are not referenced in the CLB.	None of the fatigue analyses described are reflected in LRA Table 4.3.1-1. The fatigue analyses for the BIT Nozzle at Socket Weld location contained in the WCAP-16994-P and 16995-P reports are not reflected in LRA Table 4.3.4-1. The projected 60-year cumulative usage factor (CUF), fatigue life correction factor (Fen), and overall environmentally-assisted CUF are listed in Tables 4.3.7-1 and 4.3.7-2 for Salem Units 1 and 2, respectively. Input for these tables is from WCAP-16994-P and WCAP-16995-P with a Code Edition of 1986.
Unit 1 Auxiliary Feedwater Nozzle Transition Piece	1997 – The Auxiliary Feedwater Nozzle Transition Pieces, one for each of the four feedwater nozzles, were new components included as part of the Unit 1 Steam Generator replacement project. The transition pieces (forgings) were designed to the requirements of ASME Section III, 1989 edition. This analysis is considered the governing fatigue analysis. This analysis is not referenced in the CLB.	This fatigue analysis is not reflected in LRA Table 4.3.1-1. The 1997 design report evaluated the Salem Units Auxiliary Feedwater Nozzle Transition Pieces, where the Code Edition of 1989 is reflected in LRA Table 4.3.4-1. This location was not addressed for environmentally assisted fatigue, and is not reflected in LRA Tables 4.3.7-1 and 4.3.7-2.

References for above table:

1. WCAP-12914, Rev. 1, "Structural Evaluation of Salem Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," June 1992

2. WCAP-16194, Rev. 0, "Evaluation of Pressurizer Insurge/Outsurge Transients for Salem Units 1 & 2", December 2003

3. WCAP-16994-P, Rev. 0, "Environmental Fatigue Evaluation for Salem Unit 1," January 2009

4. WCAP-16995-P, Rev. 0, "Environmental Fatigue Evaluation for Salem Unit 2," January 2009

The following discussions are in regards to the requirements for an updated ASME Code, Section III Design Specification and Code Reconciliation in accordance with ASME Code, Section III requirements for the above follow-on fatigue analyses.

For all the component locations, with the exception of the Unit 1 Auxiliary Feedwater Nozzle Transition Piece which is not part of the reactor coolant pressure boundary, listed in the previous table, the EAF evaluations were performed to address the NUREG-1801, Vol. 2, Rev. 1 (Generic Aging Lessons Learned [GALL report]) requirement to evaluate the effects of reactor water environment on fatigue, using methodologies contained in NUREG/CR-6583, "Effects of LWR Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels", February 1998, and NUREG/CR-5704, "Effects of LWR Coolant on Fatigue Design Curves of Austenitic Stainless Steels", April 1999, for the applicable materials. These NUREG reports provide a method to account for EAF by increasing the fatigue usage factor by a fatigue life correction factor, Fen. As such, the NUREG reports do not require a complete ASME Section III qualification of the components, but only a fatigue usage factor calculation. Of the above locations evaluated for EAF, only the Pressurizer Surge Nozzle Safe End to Pipe Weld and the Surge Line Hot Leg Nozzle to Pipe Weld had an existing ASME Section III fatigue evaluation.

The Pressurizer Surge Nozzle Safe End to Pipe Weld and Surge Line Hot Leg Nozzle to Pipe Weld evaluations were updated to ASME Section III from the original B31.1 design code in WCAP-12914 to address NRC Bulletin 88-11. The Code edition used was ASME Section III 1986 edition as required by Bulletin 88-11 at that time. Since the original design was to the B31.1 Code, there was no design specification. The plant specific evaluation was based on a generic approach developed by the Westinghouse Owners Group (WOG) and documented in WCAP-12639, "Westinghouse Owners Group Pressurizer Surge Line Thermal Stratification Generic Detailed Analysis Program MUHP-1091 Summary Report", June 1990, which defined surge line stratification effects during standard Westinghouse design specification transients. The stratification effects postulated for the Salem specific evaluation during the standard Westinghouse plant transient conditions were described in WCAP-12914.

Of these two locations, only the Pressurizer Surge Nozzle Safe End to Pipe Weld location was re-evaluated in 2003 in WCAP-16194. This WCAP report was a Salem-specific report that evaluated Insurge/Outsurge transients previously evaluated by the Westinghouse Owners Group (WOG) under WCAP-14950, "Mitigation and Evaluation of Pressurizer Insurge/Outsurge Transients", February 1998, which were not specifically considered in the original design analysis for the pressurizer surge nozzle safe end to pipe weld. The design specifications were not updated to include these additional details. However, these details of the insurge/outsurge and stratification effects during the design specification transients were described in WCAP-16194. The WCAP-16194 report did not provide a formal ASME Section III Code edition reconciliation between 1986 and 1989 Code editions.

The latest evaluations of these two component locations are documented in WCAP-16994-P and WCAP-16995-P, and had used the same Code edition, ASME Section III, 1986 edition, as the WCAP-12914 report, the former fatigue analysis documented in the CLB for the Pressurizer Surge Nozzle Safe End to Pipe Weld and Surge Line Hot Leg Nozzle to Pipe Weld locations.

As shown from the previous table, the RHR Accumulator Nozzle to Pipe Weld, Normal and Alternate Charging Nozzle to Pipe Weld, and BIT Nozzle at Socket Weld piping components were originally designed to the B31.1 Code, and therefore there was no explicit design specification for fatigue analysis. Since the EAF evaluations documented in WCAP-16994-P and WCAP-16995-P only required a fatigue usage factor calculation, and not a full ASME Section III Code qualification, the ASME Section III fatigue usage factors were calculated for each piping component using transients from Westinghouse systems standard specifications applicable for Westinghouse four loop plants. The transients, ASME Code methodology and criteria used for the evaluations were documented in WCAP-16994-P and WCAP-16995-P and their supporting calculations. A formal Code reconciliation was not required since the original design for these locations were based on ASA/USAS B31.1 requirements.

<u>Bullet # 4 – Environmentally-Assisted Fatigue (EAF) Analyses for each Monitored Location</u>

With the exception of the Salem Unit 1 Auxiliary Feedwater Nozzle Transition Piece, which is not part of the reactor coolant pressure boundary, each of the locations to be monitored by WESTEMS™ has been evaluated for environmentally-assisted fatigue (EAF). These locations are considered the NUREG/CR-6260 locations. The Salem detailed response to RAI 4.3-05 (Issue 1), PSEG letter LR-N10-0243, dated July 13, 2010, discusses the methodology for selecting the limiting locations.

The EAF analyses for each monitored NUREG/CR-6260 location consisted of the following general steps:

1. Prepare transfer function databases, including thermal transfer function databases, and mechanical transfer function databases using the ANSYS Finite Element Code.

2. Create WESTEMS™ analysis section number (ASN) models of the respective component nozzle to evaluate specific component locations.

3. Define nozzle moment loads as functions of temperature using the WESTEMS™ derived value functions.

4. Define transients and create transient input files (WESTEMS™ history files).

5. Perform applicable stress and fatigue analyses of pertinent ASNs using the stress and fatigue analysis methods of ASME Code, Section III, NB-3200 to determine the 60-year cumulative usage factor (CUF) using the transfer function methodology in WESTEMS™.

6. Evaluate the reactor coolant environmental effects as a multiplier and apply this multiplier to the 60-year CUF. This step was completed in four separate sub-steps as follows.

    a. Assemble the stress cycle pair information, transient stress time history, and transient temperature time history needed for the Fen calculations from the stress and fatigue analysis.

    b. Determine applicable strain rate and temperature information for each stress cycle pair from the transient stress and temperature time histories.

    c. Using the modified rate approach to calculate integrated Fen for each stress cycle pair, apply the strain rate for the pair in the Fen equation, along with the temperature and oxygen content values, to determine Fen for the positive strain rate portion of each transient pair. Note that the above monitored locations evaluated for EAF consisted of stainless steel components only, therefore, the calculations used terms and equations from NUREG/CR-5704.

    d. Calculate the overall 60-year EAF-adjusted CUF (Uen) by incorporating the Fen for each pair, and determining the cumulative Uen as the sum of the individual Uen values.

<u>Bullet # 5 – Differences between the WESTEMS™ Stress Models and Stress Models used in the Governing Fatigue Analysis of Record</u>

As discussed in response to Bullet #3 above, the current governing fatigue analysis for each of the locations monitored by WESTEMS™, with the exception of the Unit 1 Auxiliary Feedwater Nozzle Transition Piece, is the recent EAF analysis described in WCAP-16994-P and WCAP-16995-P for Salem Units 1 and 2, respectively. Each of the EAF analyses consists of an ASME Section III fatigue analysis, incorporating up-to-date transients and associated loadings, and is therefore considered the current governing fatigue analysis of record. The stress models used in these EAF analyses are the same as the stress models employed in the WESTEMS™ online monitoring tool.

For the WESTEMS™ online monitoring for the Unit 1 Auxiliary Feedwater Nozzle Transition Piece, Salem will use a monitoring model consistent with the stress model employed in the governing fatigue analysis of record.

Bullet # 6– How Transient Counting Results are Incorporated into the EAF Results

The transient counting results (i.e., current cycles) were used as a basis for the 60-year projected cycles. The current cycles, the 60-year projected cycles, and the NSSS (40-year) Design Limit for each of the design transients are listed in Tables 4.3.1-3 and 4.3.1-4.

Either the 60-year projected cycles or the bounding NSSS (40-year) Design Limit values were used as inputs into the ASME Section III 60-year cumulative usage fatigue (CUF) calculations. The results of the calculations are listed in the Column titled "60-Year Design CUF" in LRA Tables 4.3.7-1 and 4.3.7-2. The 60-year Design CUF values were multiplied by the corresponding fatigue life correction factor, Fen, to obtain the 60-year EAF-adjusted CUF, whose values are also listed in LRA Tables 4.3.7-1 and 4.3.7-2.

WESTEMS™ Benchmarking Evaluation

Bullet # 1 – Comparison between WESTEMS™ and ASME Code, Section III

Salem is currently performing a benchmarking evaluation for both the Unit 2 Pressurizer Surge Nozzle and 1.5-inch Boron Injection Tank (BIT) Safety Injection Nozzle. The Unit 2 nozzles were selected over the Unit 1 nozzles due to the higher CUF values determined in their respective EAF analysis. The benchmarking evaluation for both nozzles will compare the calculated stresses and CUF using WESTEMS™ to the same results from traditional ASME Code, Section III CUF calculations for all transient pairs representing at least 75% of the total CUF from the ASME Code, Section III calculations.

Bullet # 2 – Differences in Benchmarking Evaluation Results

Any differences between the results of the WESTEMS™ evaluation and the ASME Code, Section III CUF calculations, along with a corresponding justification for acceptability of the differences, will be provided.

The results of the benchmarking evaluation will be submitted to the NRC by January 7, 2011.

RAI 4.3-08

Background:

By letter dated July 13, 2010, the applicant responded to RAI 4.3-05. In its response to request 1, the applicant provided a discussion on the methodology used to determine the locations that required environmentally assisted fatigue analyses, consistent with NUREG/CR-6260. In its response to request 3, the applicant stated that the correction temperature, T, and transformed oxygen content parameter, O*, were set to values of zero, therefore, the F_{en} is maximized when these two terms are set equal to zero. The applicant also stated that its primary water chemistry specification for dissolved oxygen during normal operations is less than 0.005 ppm. Furthermore, a review of the Units 1 and 2 RCS quarterly dissolved oxygen data indicated that the dissolved oxygen content was less than 0.05 ppm since 2000, except for short periods of time during start-up and shutdown conditions.

Issue:

GALL AMP X.M1 states the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260, as a minimum, or propose alternatives based on plant configuration. The staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of reactor coolant environment, other than those generic locations identified in NUREG/CR-6260. The staff noted this may include locations, for example, (1) that are limiting or bounding for a particular plant-specific configuration or (2) that have calculated CUF values that are greater than those for the locations identified in NUREG/CR-6260.

The staff noted that the statement "F_{en} is maximized when these two terms are set equal to zero" is not accurate because the last term in the F_{en} expression can be less than zero (thus subtracting a negative value and providing a higher value of F_{en} and the use of T equal to zero in the second term of the F_{en} expression is not technically correct. The staff also noted that setting the transformed oxygen content parameter, O*, to a value of zero is based on the assumption that the applicant has always operated with dissolved oxygen less than 0.05 ppm since initial plant start-up. However, the applicant's response only confirmed the dissolved oxygen content for the time period since the year 2000. The staff also noted that it is not clear how much time elapses during the short periods of time during start-up and shutdown conditions when dissolved oxygen content is greater than 0.05 ppm.

Request:

1. Confirm and justify that the locations selected for environmentally assisted fatigue analyses, consistent with NUREG/CR-6260, are the most limiting and bounding for the plant. If these locations are not the most limiting and bounding for the plant, clarify the locations that require an environmentally assisted fatigue analysis and the actions that will be taken for these additional locations. If the most limiting location consists of nickel alloy, NUREG/CR-6909 methodology for nickel alloy will be used.

2. Justify the statement "F_{en} is maximized when these two terms are set equal to zero" made in response to RAI 4.3-05.

3. Clarify whether dissolved oxygen content has always been maintained less than 0.05 ppm since initial plant start-up, and provide justification to support this clarification. If not, justify why the F_{en} values provided in LRA Tables 4.3.7-1 and 4.3.7-2 do not account for these periods of time in which dissolved oxygen content was not maintained less than 0.05 ppm, including the "short periods of time during start-up and shutdown conditions."

Salem Response:

1. Salem will perform a review of design basis ASME Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 based locations that have been evaluated for the effects of the reactor coolant environment on the fatigue usage are the limiting locations for the Salem plant configuration. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. If any of the limiting locations consist of nickel alloy, NUREG/CR-6909 methodology for nickel alloy will be used in the evaluation. These additional evaluations will be performed through the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Salem LRA Appendix B, Section 3.1.1) to manage metal fatigue associated with the environmental effects on fatigue usage in accordance with 10 CFR 54.21 (c)(1)(iii). As a result of this RAI response, commitment #52 is added to LRA Table A.5, License Renewal Commitment List, as shown in Enclosure B of this letter.

2. In its response to RAI 4.3-5, Issue (3), PSEG letter LR-N10-0243, dated July 13, 2010, Salem intended to make an overall assessment of its application of Fen to low alloy steel components, specifically the Reactor Vessel Shell and Lower Head, and the Reactor Vessel Inlet and Outlet Nozzles.

   The last sentence on page 15 of 26 in the enclosure to the response to RAI 4.3-5 (Issue 3) stated "Fen is maximized when these two terms are set equal to zero". The two terms referred to in this statement were the correction temperature, T, and the transformed oxygen content parameter, O^*. These two terms are used in Equation 6.5b from NUREG/CR-6583. (Note that a typographical error was identified in the second term of the equation associated with the Salem response to RAI 4.3-5, Issue (3) during this review. It has been corrected as shown below as a deletion in **~~bold strikethrough~~** font.)

   $\mathrm{Ln}\,(\mathrm{Fen}) = 0.929 - 0.001\cancel{1}24T - 0.101 S^* T^* O^* \varepsilon'^*$

   where:

Fen	=	Fatigue Life Correction Factor
T	=	correction temperature
S^*	=	transformed sulfur content
T^*	=	transformed temperature
O^*	=	transformed dissolved oxygen content
ε'^*	=	transformed total strain rate

   Salem concurs that this statement is not accurate for all situations, particularly when a negative transformed total strain rate, ε'^*, is used, the resultant Fen value would exceed 2.532, the Fen value computed when both terms are set to zero.

   As stated in the response to RAI 4.3-5 (Issue 3), the transformed dissolved oxygen content, O^*, was set to zero since the dissolved oxygen content was assumed to be < 0.05 ppm.

Refer to our response to Item No. 3 of this RAI for further explanation of this assumption.

Salem correctly applied a zero term for transformed dissolved oxygen content, O^*, making the third term of Equation 6.5b from NUREG/CR-6583 equal to zero for the Salem-specific environmental fatigue analyses. Also, a conservative value of zero was used for the second term in Equation 6.5b. However, the statement, "Fen is maximized when these two terms are set equal to zero" is not accurate for analyses other than the Salem environmental fatigue analyses. Salem revises its response to RAI 4.3-5, Issue (3), dated July 13, 2010. The updated response is modified below with deletions to text shown as ~~strikethrough~~.

> To provide sufficient conservatism in the Fen calculations the following assumptions were used. Salem set the correction temperature, T, to a value of zero making the second term a value of zero. Salem also set the transformed oxygen content parameter, O^*, to a value of zero, making the third term a value of zero. ~~Fen is maximized when these two terms are set equal to zero.~~

3. In its response to RAI 4.3-5, Issue (3), dated July 13, 2010, Salem stated "A review of the Salem Units 1 and 2 RCS quarterly dissolved oxygen (DO) data indicated that the DO content was less than 0.05 ppm since 2000, except for short periods of time during start-up and shutdown conditions". Our clarification to this statement is as follows.

   During Modes 1 (Power Operations) and 2 (Startup), where the reactor coolant system is greater than or equal to 350°F and reactivity condition (K_{eff}) is $\geq$ 0.99, the Reactor Coolant System (RCS) DO concentrations are always less than 0.05 ppm (50 ppb), specifically, less than 0.005 ppm (5 ppb) as determined from the RCS quarterly chemistry data since 2000. The reason for the extremely low DO levels is due to the RCS environment containing a hydrogen concentration of a minimum of 25 cubic centimeters per kilogram of coolant volume (cc/kg). This condition is specified for Westinghouse pressurized water reactors to keep the oxygen level in the RCS below the limit of detection (5 ppb). Salem had this specification limit of RCS hydrogen imposed since original start-up of the units.

   For the carbon and low alloy components, Salem used Equation 6.5b from NUREG/CR-6583. The transformed temperature, T*, is set to zero when the RCS temperatures < 150°C (302°F), which negates the contribution from DO, specifically, the transformed oxygen content parameter, O^*.

   Therefore, any DO values exceeding 0.05 ppm (50 ppb) during Mode 5 (Cold Shutdown – RCS temperature < 200°F) and Mode 6 (Refueling – RCS temperature < 140°F) do not contribute to EAF due to the low RCS temperatures.

   There are possible short periods of time where the RCS DO levels can exceed 0.05 ppm, while the RCS temperatures exceed 150°C (302°F) for carbon and low alloy steel. These short periods of time are during start-up and shutdown, or specifically, Mode 3 (Hot Standby – RCS temperature $\geq$ 350°F and K_{eff} is < 0.99) and Mode 4 (Hot Shutdown – 200°F < RCS temperature < 350°F and K_{eff} is < 0.99). Salem controls RCS DO levels to $\leq$ 0.10 ppm (100 ppb) during plant startups, specifically prior to the RCS reaching 250°F (121°C). The oxygen control is attained through hydrazine addition to the primary system.

Therefore, during the time when the RCS is heating from 302°F (Mode 4) to 350°F (Mode 3), or cooling from 350°F (Mode 3) to 302°F (Mode 4), the RCS DO levels could exceed 0.05 ppm (50 ppb), but are less than or equal to 0.10 ppm (100 ppb) The negligible impact on the EAF calculations for carbon and low alloy steel components is discussed below.

The short periods of time are less than 24 hours per plant heatup and are less than 8 hours per plant cooldown. The impact of these short periods of time is described as follows. As compared to Salem Unit 1, Unit 2 has a higher projected value for heatups and cooldowns of 157 and 155, respectively for a period of 60 years. For additional conservatism, the 40-year Nuclear Steam System Supply design specification of 200 heat-ups and 200 cooldowns is multiplied by a time period of 24 hours for the heatup event and 8 hours for the Cooldown event as ((200*24 hrs + (200*8 hrs)), resulting in 6,400 hours. The projected effective full power hours for each Salem Unit is obtained by multiplying the effective full power years of 50 by 8,760 hours in a year, or 438,000 hours. Therefore, the percentage of time that the RCS temperature will be heating from 302°F to 350°F, and cooling from 350°F to 302°F is less than 1.5% of the total operating time.

To assess the overall impact of the short periods of time where the DO is greater than 0.05 ppm, but less than 0.10 ppm, the Fen is re-calculated using Equations 5.5 and 6.5b from NUREG/CR-6583, and conservative values for S^*, O^*, and ε'^* as follows:

$$\text{Ln (Fen)} = 0.929 - 0.00124T - 0.101 S^* T^* O^* \varepsilon'^*$$

T	=	25°C
S^*	=	0.015
T^*	=	176.7°C - 150°C = 26.7°C
O^*	=	LN(0.10/0.04) = 0.916
ε'^*	=	LN(0.001) = -6.908

Therefore, a Fen approximately equal to 3.171 was computed for the short periods of times where DO is greater than 0.05 ppm, but less than 0.10 ppm.

An adjusted Fen to account for these short periods of time can then be computed. For 1.5% of the operating time, the Fen is 3.171, and for the balance of the operating time (98.5%) the Fen is 2.532. Therefore, the adjusted Fen is computed from (0.015*3.171) + (0.985*2.532), which equals 2.542. To determine the impact on the EAF calculations, the following example is shown.

The highest 60-Year Design CUF for a low alloy steel component is 0.1510, and is associated with the Units 1 and 2 Reactor Vessel Inlet Nozzles as shown in LRA Tables 4.3.7-1 and 4.3.7-2, respectively. Multiplying the above adjusted Fen, 2.542, by the 60-Year Design CUF, 0.1510, yields a value of 0.3838 for the 60-Year EAF-

Adjusted CUF. This is a 0.4% increase over the value of 0.3823 as reported in LRA Tables 4.3.7-1 and 4.3.7-2.

As seen from this example, there is no appreciable impact to the Fen values for the carbon and low alloy steel components provided in LRA Tables 4.3.7-1 and 4.3.7-2, and the 60-year EAF-Adjusted CUFs for all carbon and low alloy steel component locations remain below 1.00.

For stainless steel components, Salem used Equation 13 from NUREG/CR-5704. The transformed oxygen, O*, was set to 0.260, corresponding to a DO content of < 0.05 ppm, resulting in the more conservative contribution from DO. Therefore, for stainless steel components, Salem used the conservative approach with respect to DO content, therefore the short periods of time where DO equals or exceeds 0.05 ppm is not a concern.

The Salem units have not changed their chemistry control with regards to oxygen control in the RCS when the temperature > 302°F since original plant start-up, therefore, the values observed in the past ten years (2000 to 2010) are representative of past operations. Salem will continue to and is committed to maintain its primary water chemistry, including the previously discussed limitations on DO, through the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2), which incorporates EPRI guidelines.

Enclosure B
Update to License Renewal Commitment List

As a result of this RAI response, the commitment discussed above is added to LRA Table A.5, License Renewal Commitment List as commitment number 52, as shown below. Any other actions described in this letter are not regulatory commitments and are described for the NRC staff's information:

A.5 License Renewal Commitment List

No.	Program or Topic	Commitment	UFSAR Supplement Location (LRA App. A)	Enhancement or Implementation Schedule	Source
52	Metal Fatigue of Reactor Coolant Pressure Boundary	Salem will perform a review of design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 based locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Salem plant configuration. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. If any of the limiting locations consist of nickel alloy, NUREG/CR-6909 methodology for nickel alloy will be used in the evaluation.	N/A	Prior to the period of extended operation.	Salem Letter LR-N10-0445 RAI 4.3-08