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JUL 13 2010

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

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

Subject: Response to NRC Request for Additional Information dated June 14, 2010, Related to Sections 4.3 and 4.4 of 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 REGARDING SECTIONS 4.3 AND 4.4 (TAC NOS. ME1834 AND ME1836)" dated June 14, 2010

In the referenced letter, the NRC requested additional information related to Sections 4.3 and 4.4 of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application. Enclosed is the response 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.

A141
NRR

JUL 13 2010

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

Executed on 7/13/10

Sincerely,

Paul J. Davison

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

Enclosure: Response to Request for Additional Information

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

Enclosure

Response to Request for Additional Information related to Sections 4.3 and 4.4 of the Salem Nuclear Generating Station, Units 1 and 2, License Renewal Application (LRA).

RAI 4.3-01
RAI 4.3-02
RAI 4.3-03
RAI 4.3-04
RAI 4.3-5
RAI 4.3-6
RAI 4.4.2-1
RAI 4.4.5-1
RAI 4.4.5-2

Note: For clarity, portions of the original LRA text are repeated in this Enclosure. Added text is shown in ***Bold Italics***, and deletions are shown with strikethrough text.

RAI 4.3-01

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

License renewal application (LRA) Section 4.3 states that, as of December 31, 2007, Salem Nuclear Generating Station (SNGS), Units 1 and 2 have been operational for 31.4 and 27.8 calendar years, respectively. This included non-operational periods of 2.9 and 2.2 years for SNGS, Units 1 and 2, respectively. LRA Section 4.3 further states that the average rate of cycle occurrences was determined from the cumulative number of cycle occurrences and 28.5 and 25.6 years of past operation for SNGS Units 1 and 2, respectively. However, LRA Section 4.3 does not provide sufficient information for the staff to confirm that cycle counting has been performed from the plant start-up and during the entire period of past operation prior to December 31, 2007.

Request:

Clarify whether the cycle counting for the design basis transients at SNGS Units 1 and 2 has been performed during the entire period of past operation (i.e., over the entire time of operation since the initial startups of the units, including times during heatup and cooldown conditions but not including hot or cold shutdown conditions or hot standby conditions), or whether any unmonitored periods exist during plant operation.

PSEG Response:

In preparation of the Time-Limited Aging Analyses (TLAAs), Salem reviewed the cycle count information from the current cycle counting program as of December 31, 2007. The cycle counting program was current up through December 31, 2007. An independent review was conducted of past plant historical documents, including Licensee Event Reports (LERs), Monthly Operating Reports (MORs), and the Units' computer-based data archive system to establish the cycle counts for the design basis transients as of December 31, 2007. This independent review confirmed cycle counting for the design basis transients at Salem Units 1 and 2 and confirmed there were no unmonitored periods during the entire period of past operation. This review included the entire time of operation since the initial startups of the Salem units, including times during heatup and cooldown conditions, and hot standby conditions, but not including hot shutdown (Mode 4) or cold shutdown (Mode 5) conditions. Both Salem Units 1 and 2 had incurred the Hot Standby transients in the past as shown in Salem LRA Tables 4.3.1-3, "Design Transients and 60-Year Projections for NSSS Class A and Class 1 Components at Salem Unit 1" and 4.3.1-4, "Design Transients and 60-Year Projections for NSSS Class A and Class 1 Components at Salem Unit 2". As noted in the two tables, the number of cycles of Hot Standby Operations remains constant following implementation of digital feedwater control at both Salem units.

For each of the design basis transients listed in LRA Table 4.3.1-3 and Table 4.3.1-4, Salem had used the larger of the two values (2007 annual cyclic data report or independent review of plant historical information) for current cycles, to determine the respective Salem Unit 60-year projections.

RAI 4.3-02

Background

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue

LRA Table 4.3.1-2 lists the limiting design basis number of occurrences for 40 years for nuclear steam supply system (NSSS) Class A and Class 1 components at SNGS Units 1 and 2. However, LRA Section 4.3.1 does not reference the design basis documents that confirm the limiting design basis number of occurrences provided in LRA Table 4.3.1-2.

Request

Clarify which current licensing basis (CLB) documents or design basis documents provide CLB or design basis transient cycle limits for the transients that are listed in LRA Table 4.3.1-2 for NSSS Class A and Class 1 components at SNGS.

PSEG Response:

The following reference table provides a current licensing basis or design basis document reference for each of the transients listed in Salem LRA Table 4.3.1-2, "Design Transient Cycles for NSSS Class A and Class 1 Components at Salem Units 1 and 2".

Design Transient Cycles for NSSS Class A and Class 1 Components at Salem Units 1 and 2		
Transient Description	**Limiting Design Basis Number of Occurrences for 40 Years**	**Current Licensing Basis or Design Basis Reference**
Normal Condition Transients:		
Plant Heat Up @ ≤ 100 °F/hr	200	**(1), (2), (3)**
Plant Cooldown @ ≤ 100 °F/hr	200	**(1), (2), (3)**
Unit Loading @ 5%/min	13,200	**(2)**
Unit Unloading @ 5%/min	13,200	**(2)**
Step Load Increase of 10% of Full Power	2,000	**(2), (3)**
Step Load Decrease of 10% of Full Power	2,000	**(2), (3)**
Large Step Load Decrease (50%) with Steam Dump	200	**(1), (2), (3)**
Hot Standby Operations	2,000	**(4)**
Upset Condition Transients:		
Loss of Load	80	**(1), (2), (3)**
Loss of Power	40	**(1), (2), (3)**
Loss of Flow	80	**(1), (2), (3)**
Reactor Trip from Full Power	400	**(1), (2), (3)**
Inadvertent Auxiliary Spray to Pressurizer	10 (original) 5 (current)	**(4) (original)** **(5) (current)**
Inadvertent Safety Injection	50	**(6)**
Test Condition Transients:		
Turbine Roll Test	10	**(1), (2), (3)**
Primary Side Hydrostatic Test	5	**(1), (2), (3)**
Secondary Hydrostatic Test	5	**(1), (2), (3)**
Primary Side Leak Test	50	**(1), (2), (3)**

(1) Salem Units 1 and 2 Technical Specifications, Table 5.7-1, "Component Cyclic or Transient Limits"
(2) Salem UFSAR Table 5.2-10a, "Design Thermal and Loading Cycles*, Model F SG – Unit 1", Rev. 24
(3) Salem UFSAR Table 5.2-10, "Design Thermal and Loading Cycles*, AREVA NP Model 61/19T SG – Unit 2", Rev. 24
(4) WCAP-12914, "Structural Evaluation of Salem Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification", Rev. 1
(5) PSEG Calculation 3SC-013, "Salem Unit 1 & 2 – NRC Bulletin 88-08 Evaluation of Aux. Spray Line", Rev. 0
(6) Salem Safety Evaluation SGS/M-SE-006, "Safety Injection Transients, 1 and ½ Inch Injection Nozzles – Reactor Coolant System, No. 1 Unit", Rev. 0, 2/9/1977

Additionally, Salem had incorrectly listed a value of 10 cycles for the NSSS Design Limit for the Inadvertent Auxiliary Spray to Pressurizer transient in three LRA tables. As correctly discussed in LRA Section 4.3.4.1, this value was reduced to 5 cycles in a 1999 fatigue analysis that supported PSEG's response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems", June 22, 1988, with supplements in 1988 and 1989. For both Salem units, the respective 60-year projected cycles for this design basis transient remain less than the previous design value of 5 cycles. Therefore, Salem Table 4.3.1-2 is revised as follows.

Table 4.3.1-2 Design Transient Cycles for NSSS Class A and Class 1 Components at Salem Units 1 and 2

Transient Description	Limiting Design Basis Number of Occurrences for 40 Years
Upset Condition Transients:	
Inadvertent Auxiliary Spray to Pressurizer	~~10~~ ***5***

Also as a result of the correction to the number of design cycles for the Inadvertent Auxiliary Spray to Pressurizer transient, Salem Table 4.3.1-3 is also revised as follows.

Table 4.3.1-3 Design Transients and 60-Year Projections for NSSS Class A and Class 1 Components at Salem Unit 1

Design Transient	Current Cycles	60-Year Projected Cycles	NSSS Design Limit
Upset Condition Transients:			
Inadvertent Auxiliary Spray to Pressurizer	1	2	~~10~~ ***5***

Finally, as a result of the correction to the number of design cycles for the Inadvertent Auxiliary Spray to Pressurizer transient, Salem Table 4.3.1-4 is revised as follows.

Table 4.3.1-4 Design Transients and 60-Year Projections for NSSS Class A and Class 1 Components at Salem Unit 2

Design Transient	Current Cycles	60-Year Projected Cycles	NSSS Design Limit
Upset Condition Transients:			
Inadvertent Auxiliary Spray to Pressurizer	1	3	~~10~~ ***5***

RAI 4.3-03

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.4.4 states that the 60-year projected cycles applicable to the SNGS Unit 1 steam generator (SG) manway studs are bounded by the cycles used in 40-year fatigue analysis. However, LRA does not provide sufficient information to confirm this assertion.

Request:

Identify the transients that were used in 40-year fatigue analyses of the SG manway studs. Provide the 60-year cycle projections for any transients that are applicable to fatigue analyses of the SG manway studs but that are not within the scope of the transients that are listed in LRA Table 4.3.1-3 and 4.3.1-4 (which list the 60-year transient cycle projections for SNGS Units 1 and 2, respectively).

PSEG Response:

As discussed in Salem LRA Section 4.3.4.4, Salem Unit 1 Steam Generator Primary Manway Studs, Westinghouse conducted a series of tests to qualify their life for forty (40) years. The primary manway stud fatigue qualification tests were performed for Westinghouse Model F Steam Generators in accordance with ASME Code, Section III, Appendix II, 1989 edition. The Time-Limited Aging Analysis (TLAA) discussed in LRA Section 4.3.4.4 is only applicable to Salem Unit 1, therefore, only LRA Table 4.3.1-3, "Design Transients and 60-Year Projections for NSSS Class A and Class 1 Components at Salem Unit 1", is addressed in this response.

The primary manway stud fatigue qualification tests were conducted with test cycles specifically developed for the duty cycles of another utility's Model F Steam Generator and were performed to bound other Model F steam generators, including Salem Unit 1. The 40-year design transients used to determine the test parameters for the primary manway stud fatigue qualification tests were from the general design specifications for the Westinghouse Model F Steam Generator. Since the transients used for the fatigue qualification tests considered a larger population of steam generators, "additional transients" were considered in determining the test parameters. All of the 40-year design transients in the general design specification for Model F Steam Generators were determined to bound the corresponding 40-year design transients for the Salem Unit 1 Steam Generators. Although these "additional transients" were a basis for the primary manway stud fatigue qualification test parameters, they were not considered for 60-year cycle projections at Salem Unit 1 since they are not part of the current licensing basis for the Salem Unit 1 Steam Generators as presented in Salem UFSAR Table 5.2-10a.

The 40-year design transients for the Salem Unit 1 Model F Steam Generators are bounded by those presented in Salem LRA Table 4.3.1-3, NSSS Design Limit transients. There are no other 40-year design transients that are applicable to the Salem Unit 1 Model F Steam Generator Primary Manway Studs fatigue analysis that are not listed in Salem LRA Table 4.3.1-3. Therefore, the 60-year cycle projections contained in Salem LRA Table 4.3.1-3 are bounded by the test parameters used for the primary manway stud fatigue qualification testing.

Although a cumulative usage factor (CUF) was not calculated for the primary manway stud fatigue evaluation, dye penetrant examinations of the studs conducted after the fatigue tests were completed to confirm no surface cracking had occurred, and the stress-strain behavior of the studs monitored throughout the test indicated no strength degradation. Based on these results, the CUF was determined by Westinghouse to be less than 1.0. Since the 60-year cycle projections were bounded by the test parameters used for the primary manway stud fatigue qualification testing, the 60-year projected CUF is also less than 1.0.

RAI 4.3-04

Background

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue

LRA Section 4.3.4.3 states that the thermal stratification loads are managed by LRA aging management program (AMP) B3.1.1 "Metal Fatigue of Reactor Coolant Pressure Boundary Program," where the number of auxiliary feedwater flow operational hours are tracked and compared to the design limit of 12,000 hours. However LRA AMP B3.1.1 is described in the LRA as consistent with Generic Aging Lessons Learned (GALL) AMP X.M1, "Metal Fatigue of the Reactor Coolant Pressure Boundary," which pertains to cycle counting of occurrence of design basis transients, not to the tracking of amassed time of operation (in seconds, minutes, hours, etc.). The LRA does not provide sufficient information for the staff to determine how LRA AMP B3.1.1 is designed to track amassed hours against an hourly design limit of 12,000 hours when the program is designed to track cumulative number of design basis transient occurrences that occur at the Salem unit facilities (i.e., to perform cycle tracking) in order to comply with technical specification 5.7.1 and other design basis requirements.

Request

Justify why the tracking of hours for the auxiliary feedwater nozzle components has not been identified as an enhancement to the "detection of aging effects" program element in GALL AMP X.M1 and why it is valid to use the program as a basis for tracking the cumulative number of hours for which the auxiliary feedwater nozzles have been in operation, when the program is designed to track the number of design basis transient cycle occurrences. If it is valid to track the auxiliary feedwater nozzles in this manner, the LRA description of the AMP should be amended to identify this as an enhancement to the following program elements in GALL AMP X.M1, with explanations and justifications on the enhancements: (1) "detection of aging effects" program element, with an explanation/justification on how the LRA AMP B3.1.1 is different from GALL AMP X.M1 to permit tracking the cumulative number of hours that these components have been in operation as opposed to tracking discrete occurrences of transients; (2) "acceptance criteria" program element, with an explanation/justification of the action limit that will be used to take appropriate corrective actions if the number of tracked hours is determined to encroach on the 12,000 hour design limit; (3) "monitoring and trending" program element, with an explanation/justification on how the tracking of the cumulative hours of operation will be trended against the defined action limit on hourly tracking; and (4) the "corrective actions" program element, with an explanation/justification on the corrective actions that will be applied to the auxiliary feedwater nozzle components if the monitoring and trending of cumulative hours in operation reaches the established action limit on cumulative hour tracking.

PSEG Response:

Salem has reviewed its Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Salem LRA Appendix B, Section B.3.1.1) and the impact of manually tracking hours of Auxiliary Feedwater (AFW) Pump operation. In lieu of manually tracking hours of AFW Pump operation during the period of extended operation, Salem will use WESTEMS™ to automatically compute cumulative usage for the Unit 1 Steam Generator Feedwater Nozzle transition piece, based on input from local plant instrumentation. A limit will be determined for cumulative usage, based on AFW pump operation, at the transition piece as opposed to tracking the number of auxiliary feedwater flow operational hours and comparison to the design limit of 12,000 hours. As discussed in Salem LRA Section 4.3.4.3, all design basis transients considered in the original analysis remain the same, and are monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program. The Hot Standby transient was replaced with the thermal stratification loads which are caused by AFW pump operation. Therefore, for this component, Salem Metal Fatigue of Reactor Coolant Pressure Boundary aging management program will calculate the cumulative usage based on design basis transients and the stratification loads to ensure the overall cumulative usage is less than 1.00. If the fatigue usage for this location has an unanticipated increase or if the cumulative usage is approaching its limit, i.e., 80% of its design limit, the Corrective Action Program (CAP) is initiated to evaluate the condition and determine the corrective action.

Salem revises its Analysis and Disposition discussions under Salem LRA Section 4.3.4.3, Salem Unit 1 Steam Generator Feedwater Nozzle Transition Piece as follows.

Analysis

The ~~thermal stratification loads are~~ ***cumulative usage for the Unit 1 Steam Generator Feedwater Nozzle Transition Piece, is based on design basis transients and thermal stratification loads. The cumulative usage will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary program, Salem LRA Appendix B, Section B.3.1.1 and implemented as Enhancement 2. The program will monitor cumulative usage and prevent exceeding its*** ~~number of Auxiliary Feedwater Flow operational hours will be tracked and compared to the design limit of 12,000 hours~~ ***design limit of 1.00.*** ~~Additionally, the 60-year projected numbers of applicable transients for the transition piece design analysis are within the individual transient design values. The Metal Fatigue of Reactor Coolant Pressure Boundary program monitors the transients to compare them to the design values.~~

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation by the Metal Fatigue of Reactor Coolant Pressure Boundary program (B.3.1.1), which monitors transient cycles ***and uses a software program, WESTEMS™, to compute cumulative usage for select locations,*** to assure they do not exceed their design limits, validating the assumptions used in these evaluations.

Additionally, the third paragraph of Salem LRA Appendix A, Section A.4.3.4.3 is revised as follows:

A.4.3.4.3 Salem Unit 1 Steam Generator Feedwater Nozzle Transition Piece

The ~~thermal stratification loads~~ ***cumulative usage for the Unit 1 Steam Generator Feedwater Nozzle Transition Piece, is based on design basis transients and thermal stratification loads. The cumulative usage*** will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program, where the ***program will monitor cumulative usage and prevent exceeding its*** ~~number of Auxiliary Feedwater Flow operational hours will be tracked and compared to the design limit of 12,000 hours.~~ ***design limit of 1.00.*** ~~Additionally, the 60-year projected numbers of applicable transients for the transition piece design analysis are within the individual transient design values. The Metal Fatigue of Reactor Coolant Pressure Boundary program monitors the transients to compare them to the design values.~~

RAI 4.3-5

Background

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue (1)

LRA Section 4.3.7 states that, using plant specific design fatigue results, the applicant identified the plant-specific components for the NUREG/CR-6260 sample locations and the SNGS locations that bound those of NUREG/CR-6260 sample locations (LRA Tables 4.3.7-1 and 4.3.7-2, for SNGS Units 1 and 2, respectively). Further, the applicant performed environmentally-assisted fatigue (EAF) calculations for these SNGS locations to evaluate the effects of the reactor coolant system environment on fatigue life. However, the LRA does not provide sufficient information on the methodology used in determining those SNGS locations that bound those of NUREG/CR-6260 sample locations and the basis for performing EAF calculations for these locations in place of EAF calculations for identified NUREG/CR-6260 plant-specific components.

Request (1)

Explain the methodology used in determining the plant-specific, limiting locations within the boundary of the applicable NUREG/CR-6260 component locations.

PSEG Response:

The Salem plant-specific, limiting locations that bound the applicable NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components", March 1995, component locations for an Older Vintage Westinghouse plant are listed in Tables 4.3.7-1, "Salem Unit 1 60-Year Environmentally-Assisted Fatigue Results" and 4.3.7-2, "Salem Unit 2 60-Year Environmentally-Assisted Fatigue Results".

The methodology used for determining each of the six (6) Salem plant-specific locations was as follows.

1. Reactor Vessel Shell and Lower Head

   Based on review of the reactor vessel fatigue analyses, and guidance presented in Section 5.5.1 of NUREG/CR-6260, Salem selected the Core Support Guide Welds as the limiting component for the Reactor Vessel Shell and Lower Head. The specific controlling fatigue location was the outer corners of the weld that connects core support guide to the reactor vessel inner wall.

2. Reactor Vessel Inlet and Outlet Nozzles

   Based on review of the reactor vessel fatigue analyses, and guidance presented in Section 5.5.2 of NUREG/CR-6260, Salem selected the Reactor Vessel Inlet and Outlet Nozzles as limiting components. For both the Inlet and Outlet Nozzles, the specific controlling fatigue location was the outside surface of the nozzle-to-shell juncture.

3. Pressurizer Surge Line (including hot leg and pressurizer nozzles)

   The Salem pressurizer surge line nozzles were evaluated for fatigue in WCAP-12913, "Structural Evaluation of Salem Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification", Rev. 1, dated June 1992, for the effects of surge line thermal stratification and plant-specific transients. A later Salem fatigue analysis for the pressurizer lower head and surge nozzle was performed in WCAP-16194, "Evaluation of Pressurizer Insurge/Outsurge Transients for Salem Units 1 & 2", Rev. 0, dated December 2003.

   Based on review of these fatigue analyses, and guidance presented in Section 5.5.3 of NUREG/CR-6260, Salem selected the surge line hot leg nozzle as a limiting component for the Pressurizer Surge Line.

   Due to the relatively high fatigue of the pressurizer surge and hot leg nozzles, Salem developed detailed models (i.e., finite element analysis) and applied a stress analysis for each of the nozzles and their connections to determine their respective exact limiting location. The specific controlling fatigue location was the stainless steel weld that connects the hot leg surge nozzle safe end to the surge line piping.

4. Reactor Coolant System (RCS) Piping Charging System Nozzles

   Based on guidance presented in Section 5.5.4 of NUREG/CR-6260, Salem selected both the normal and alternate charging nozzles, which are identical 3-inch nozzles, as the limiting fatigue locations for this NUREG/CR-6260 location. Salem developed a detailed model (i.e., finite element analysis) of the 3-inch charging nozzles and applied a stress analysis for the nozzles and their connections to determine the exact limiting location. The specific controlling fatigue location was the weld that connects the nozzle to the charging line piping.

5. Reactor Coolant System (RCS) Piping System Safety Injection Nozzles

   Based on guidance presented in Section 5.5.5 of NUREG/CR-6260, Salem reviewed the Safety Injection System nozzles connected to the RCS cold leg. These nozzles include the following three (3) types of RCS cold leg nozzles; 10-inch accumulator, 1.5-inch Boron Injection Tank (BIT), and 6-inch safety injection return on the hot legs. Based on a review of the severity of the piping loads and Salem-specific transients, the 1.5-inch BIT nozzles were selected to represent this NUREG/CR-6260 location.

   Salem developed a detailed model (i.e., finite element analysis) of the 1.5-inch BIT nozzles and applied a stress analysis for each of the nozzles and their connections to determine their respective exact limiting location. The specific controlling fatigue

location was the BIT piping region at the socket weld that connects the nozzle to the safety injection line piping.

6. Residual Heat Removal (RHR) System Class 1 Piping

   Based on guidance presented in Section 5.5.6 of NUREG/CR-6260, Salem reviewed the RHR Class 1 piping, specifically, the letdown path and return path to the RCS primary loop. Based on a review of the transient loads in the letdown and return paths, Salem selected the 10-inch accumulator/RHR injection cold leg nozzles as the limiting fatigue location for this NUREG/CR-6260 location. Salem developed a detailed model (i.e., finite element analysis) and applied a stress analysis for these nozzles and their connections to determine the exact limiting location. The specific controlling fatigue location was the weld that connects the accumulator nozzle to the RHR line piping.

Issue (2)

LRA Section 4.3.7 does not provide sufficient information on the assumptions and the basis for assumptions used in the 60-year cumulative usage factor (CUF) calculations for the NUREG/CR-6260 sample locations.

Request (2)

Identify all assumptions used in the 60-year CUF calculations for the NUREG/CR-6260 sample locations. Provide the basis why the assumptions applied to the 60-year CUF calculations are considered to be capable of yielding sufficiently conservative CUFs for application to SNGS EAF calculations.

PSEG Response:

Salem prepared the Environmental Assisted Fatigue (EAF) calculations by first generating the 60-year Cumulative Usage Factor (CUF) calculations for the six (6) NUREG/CR-6260 sample locations as listed in Tables 4.3.7-1 and 4.3.7-2, and then applying the Fatigue Life Correction Factor, Fen.

The assumptions used in generation of the 60-year CUF calculations for the NUREG/CR-6260 sample locations were that the 40-Year Nuclear Steam Supply System (NSSS) transient design cycles and auxiliary transient design cycles, or their respective 60-Year projected number of cycles would bound the actual number of cycles experienced by the Salem Units 1 and 2 for each of the transients through the period of extended operation. Any NSSS transients whose 40-year NSSS transient design cycles bound the corresponding lower 60-year projected number of cycles, provide inputs to yield sufficiently conservative CUFs for application to the Salem Units 1 and 2 EAF calculations.

To validate the bases for the assumptions, Salem will implement the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Salem LRA Appendix B, Section B.3.1.1) to monitor transients, and use WESTEMS™ to compute cumulative fatigue at select NUREG/CR-6260 sample locations to ensure that the 60-year CUF values for all of the

NUREG/CR-6260 sample locations remain less than the design limit. Implementation of this program is a commitment in the Salem LRA Appendix A, Section A.5 (Item 47).

Issue (3)

LRA Section 4.3.7 does not provide sufficient information on the basis for assumptions used in the environmental fatigue multipliers (Fen) calculations for the NUREG/CR-6260 sample locations.

Request (3)

Identify all assumptions used (e.g., sulfur content, dissolved oxygen, temperature, strain rate) in the Fen calculations for the NUREG/CR-6260 sample locations. Provide your basis why the assumptions applied to the Fen calculations are considered to be capable of yielding sufficiently conservative Fen factors for application to SNGS EAF calculations.

PSEG Response:

Salem used the methodology contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels", March 1998, and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels", April 1999, to evaluate environmental effects on carbon and low alloy steels, and austenitic stainless steels, respectively. The assumptions (e.g., sulfur content, dissolved oxygen, temperature, strain rate) used and bases in the Fen calculations for the NUREG/CR-6260 sample locations are discussed below.

For low alloy steel components, the controlling fatigue location associated with the NUREG/CR-6260 sample locations of the Reactor Vessel Shell and Lower Head, and the Reactor Vessel Inlet and Outlet Nozzles, Salem used Equation 6.5b from NUREG/CR-6583:

$$\text{Ln (Fen)} = 0.929 - 0.001124T - 0.101S^{*}\,T^{*}\,O^{*}\,\varepsilon'^{*}$$

where:

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

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.

Therefore, $Fen = \exp(0.929 - 0.001124(0) - 0.101S^{*} T^{*} (0) \varepsilon'^{*})$

$Fen = 2.532$ for the Reactor Vessel low alloy steel components

For austenitic stainless steels, the controlling fatigue location associated with the NUREG/CR-6260 sample locations of the Pressurizer Surge Line, the RCS Piping Charging System Nozzles, the RCS Piping System Safety Injection Nozzles, and the RHR System Class 1 Piping, Salem used the following equation from NUREG/CR-5704:

$Fen = \exp(0.935 - T^{*} O^{*} \varepsilon'^{*})$

Below are the assumptions used to develop the values for Fen.

$T^{*} = 0$ for conditions where $T < 200°C$

$T^{*} = 1.0$ for conditions where $T \geq 200°C$

$O^{*} = 0.260$ for conditions where Dissolved Oxygen (DO) < 0.05 ppm

$O^{*} = 0.172$ for conditions where Dissolved Oxygen (DO) $\geq$ 0.05 ppm

For Salem, O^{*} was determined to be a value of 0.260 since the DO was assumed to be < 0.05 ppm. This assumption is based on the Salem primary water chemistry specification for DO during normal operations of less than 5 ppb (0.005 ppm). A review of the Salem Units 1 and 2 RCS quarterly dissolved oxygen 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. The Salem primary water chemistry specifications require the DO content to be reduced to less than 100 ppb (0.10 ppm) prior to heating up above 250°F (121°C).

For the transformed strain rate, ε'^{*}, the following three values could be used, depending upon the strain rate, ε', at the location.

$\varepsilon'^{*} = 0$ for conditions where $\varepsilon' > 0.4\%/sec$

$\varepsilon'^{*} = \ln(\varepsilon'/0.4)$ for conditions where $0.0004 \leq \varepsilon' \leq 0.4\%/sec$

$\varepsilon'^{*} = \ln(0.0004/0.4)$ for conditions where $\varepsilon' < 0.0004\%/sec$

To incorporate the strain rate into the austenitic stainless steel Fen calculations, a detailed integrated method, known as the modified rate approach, was used where the Fen is integrated over the strain range for the tensile strain producing cycle of the transient pair.

$$Fen = \frac{\sum Fen_i \Delta\varepsilon_i}{\sum \Delta\varepsilon_i}$$

where:

Fen_i = Fen computed for time interval I, based on $\varepsilon' = 100\Delta\varepsilon' / \Delta t_i$, and transformed parameters T^*, O^*, and ε^* computed for the interval
$\Delta\varepsilon_i$ = change in strain for the time interval I, $(\sigma_i - \sigma_{i-1}) / E$
σ_i = stress intensity for time i
σ_{i-1} = stress intensity for time i-1
Δt_i = change in time for time interval I, $\Delta t = t_i - t_{i-1}$
E = Young's Modulus

Transient total stress time histories were used to determine the corresponding strain rates of the tensile producing portion of the stress cycle for the different fatigue pairs for all of the applicable analyzed transients. A Fatigue Life Correction Factor, Fen, was calculated for all fatigue pairs, and then used in the above equation.

Issue (4):

LRA Section 4.3.7 does not indicate the material at each of the critical fatigue locations identified in the section.

Request (4):

Clarify whether any of the critical fatigue locations include nickel alloys. If so, identify the source and justification of the Fen formula used for the nickel alloy calculations.

PSEG Response:

None of the six (6) NUREG/CR-6260 critical fatigue locations included nickel alloy materials. The critical fatigue locations associated with the Reactor Vessel Shell and Lower Head, and the critical fatigue locations associated with the Reactor Vessel Inlet and Outlet Nozzles are constructed of low alloy steel. The critical fatigue locations associated with the Pressurizer Surge Line (including hot leg and pressurizer nozzles) are constructed of stainless steel. The critical fatigue locations associated with the RCS Piping Charging System Nozzles (3-inch normal and alternate charging nozzles) are constructed of stainless steel. The critical fatigue locations associated with the RCS Piping System Safety Injection Nozzles (1.5-inch BIT nozzles) are constructed of stainless steel. The critical fatigue locations associated with the RHR System Class 1 Piping (10-inch accumulator/RHR injection cold leg nozzle) are constructed of stainless steel. Therefore, no justification of the F_{en} formula is required since there are no critical fatigue locations for determining the affects of environmental fatigue which include nickel alloy material.

Issue (5):

The locations identified and analyzed in NUREG/CR-6260 include typical limiting locations but do not consider all plant-specific components and configurations.

Request (5):

Clarify whether any other plant-specific locations at Salem are more limiting than those identified in NUREG/CR-6260. If other Salem plant-specific locations exceed those from NUREG/CR-6260, provide EAF calculations for those locations.

PSEG Response:

The sample locations prescribed in NUREG-1800 and the Salem plant-specific components contained in the Salem LRA Tables 4.3.7-1, "Salem Unit 1 60-Year Environmentally-Assisted Fatigue Results" and 4.3.7-2, "Salem Unit 2 60-Year Environmentally-Assisted Fatigue Results" are the same for the purpose of compliance with the GALL requirements. The sample locations also comply with staff recommendations contained in NUREG-1801, Vol. 2, Rev. 1 for evaluation and monitoring of NUREG/CR-6260 components for the effects of reactor coolant environment on component fatigue life, and the selection of the limiting locations. Selection of the locations compliant with NUREG/CR-6260 and the determination of the limiting locations is presented in response to Request (1) of this RAI. Since the location selection is compliant with NUREG/CR-6260 and the limiting locations have been determined and evaluated, no other Salem plant-specific locations are required to be identified and evaluated for EAF.

RAI 4.3-6

Background

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue

LRA Section 4.3.7 states that the fatigue analyses for the NUREG/CR-6260 sample locations have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). However, LRA Section B3.1.1 states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to address the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant, as identified in NUREG/CR-6260. Therefore, it is not clear whether the applicant has chosen to use a more conservative approach and manage the effects of aging on the NUREG/CR-6260 sample locations intended functions in accordance with 10 CFR 54.21(c)(1)(iii) for the period of extended operation using the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

Request

Clarify how LRA AMP B3.1.1 "Metal Fatigue of Reactor Coolant Pressure Boundary Program," will address the effects of the reactor coolant environment on the critical components identified in NUREG/CR-6260, and whether the effects of aging on all/any NUREG/CR-6260 sample locations intended functions will be managed in accordance with 10 CFR 54.21(c)(1)(iii) for the period of extended operation using LRA AMP B3.1.1. If so, describe plans to clarify LRA Section 4.3.7 to indicate for which NUREG/CR-6260 sample locations the effects of aging will be managed in accordance with 10 CFR 54.21(c)(1)(iii).

PSEG Response:

As discussed in Salem LRA Appendix B, Section B.3.1.1, Enhancement Number 3 to the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program addresses the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260. Site procedures will be revised to implement this enhancement, specifically, a periodic fatigue monitoring report will address the effects of the reactor coolant environment for each of the six (6) NUREG/CR-6260 locations discussed in Salem LRA Section 4.3.7, "Environmentally-Assisted Fatigue Analyses".

Since Salem will use the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Salem LRA Appendix B, Section B.3.1.1) to update cumulative usage values for the six (6) NUREG/CR-6260 locations via a periodic fatigue monitoring report, which were projected to the end of the period of extended operation in LRA Section 4.3.7, the last paragraph on LRA page 4-43 is revised as follows:

Tables 4.3.7-1 and 4.3.7-2 summarize the locations where EAF results were calculated as well as the results for Salem Units 1 and 2, respectively. The evaluations showed that no cumulative usage factors with environmental penalties exceed 1.0 for 60 years of service. ***The Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (B.3.1.1) will be used to manage the aging effects of environmentally assisted fatigue for the components in Salem LRA Tables 4.3.7-1 and 4.3.7-2.***

Additionally, the Disposition on LRA page 4-45 is revised as follows:

Disposition: ~~**Revision, 10 CFR 54.21(c)(1)(ii)** – The analyses have been projected to the end of the period of extended operation.~~ ***Aging Management, 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) due to environmentally assisted fatigue will be adequately managed for the period of extended operation by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (B.3.1.1). This program monitors transient cycles and uses a software program, WESTEMS™, to compute cumulative usage at select locations to assure the components in Salem LRA Tables 4.3.7-1 and 4.3.7-2 do not exceed their design limits.***

In addition, the last two paragraphs of Salem LRA Appendix A, Section A.4.3.7 are revised as follows:

A.4.3.7 Environmentally-Assisted Fatigue Analyses

The evaluations showed that no cumulative usage factors with environmental penalties exceeded 1.0 for 60 years of service for the identified plant-specific locations. ***The Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (B.3.1.1) will be used to manage the aging effects of environmentally assisted fatigue for the components in Salem LRA Tables 4.3.7-1 and 4.3.7-2.***

~~The analyses are projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).~~ ***The Metal Fatigue of Reactor Coolant Pressure Boundary program will manage the effects of aging due to environmentally assisted fatigue in accordance with 10 CFR 54.21(c)(1)(iii).***

As a result of changing the Disposition from **10 CFR 54.21(c)(1)(ii) to 10 CFR 54.21(c)(1)(iii) in** Salem LRA Section 4.3.7, LRA Table 4.1-1 is revised as follows.

Table 4.1-1 Time-Limited Aging Analyses Applicable to Salem			
TLAA Category	**Description**	**Disposition Method(s)**	**LRA Section**
2.	**Metal Fatigue of Piping and Components**		**4.3**
	Environmentally-Assisted Fatigue Analyses	~~§54.21(c)(1)(ii)~~ ***§54.21(c)(1)(iii)***	4.3.7

RAI 4.4.2-1

Background:

LRA Section 4.4.2 discusses reactor coolant pump (RCP) flywheel fatigue crack growth analyses and states that Westinghouse Topical Report WCAP-14535A, "RCP Flywheel Inspection Elimination" includes a fatigue flaw growth analyses that has been identified as a time-limited aging analysis (TLAA). The LRA also states that the purpose of the report was to provide an engineering basis for elimination of RCP flywheel inservice inspection requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants. The LRA concludes that RCP flywheels will maintain their structural integrity during the period of the extended operation because the maximum number of start-stop cycles projected for 60 years (e.g., 661 start-stop cycles for Unit 1 and 703 start-stop cycles for Unit 2) have been demonstrated to be bounded by the 6,000 start-stop cycles limit assumed in the WCAP-14535A fatigue flaw growth analysis.

Issue:

The U.S. Nuclear Regulatory Commission (NRC or the staff) endorsed WCAP-14535 in a safety evaluation (SE) dated September 12, 1996 (ADAMS Legacy Library Accession #9609230010). In the conclusion section of the SE (Section 4.0), the staff concluded that that the inspections of the flywheels should be performed even if all of the recommendations of Regulatory Guide (RG) 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity" [August 1976], were met, and that the inspections of the RCP flywheels should not be completely eliminated. To be consistent with this SE position, the staff concluded that licensees should conduct either qualified UT or surfaces examinations of their flywheels once every ten years. It is not evident from the TLAA discussion whether the applicant intends to continue the inservice inspection (ISI) examinations of the RPC flywheels during the period of extended operation consistent with position taken in the staff's SE of September 12, 1996, or whether the applicant is proposing to discontinue the ISI examinations of the RCP flywheels during the period of extended operation.

Request:

Clarify whether the safety basis in the TLAA for the RCP flywheels is being used to justify elimination of the RCP flywheel examinations altogether, or whether the applicant intends to continue the ISI examinations of the RCP flywheels consistent the NRC's SE on WCAP-14535, dated September 12, 1996. If ISI examinations will be performed during the period of extended operation, clarify what type of examinations will be performed on the RCP flywheels during the period of extended operation and the frequency that will be used for the examinations. Justify the examination method and frequency that will be used for the ISI examinations during the period of extended operation. Otherwise, justify your basis for discontinuing the ISI examinations of the RCP flywheels if ISI examinations will be discontinued during the period of extended operation.

PSEG Response:

This RAI was previously addressed in RAI 4.4.2-1 as part of PSEG letter LR-N10-0111 to NRC dated 4/20/2010. Per PSEG discussions with the NRC License Renewal Project Manager on 7/8/2010, no further information is required for this RAI.

RAI 4.4.5-1

Background:

LRA Section 4.4.5 discusses Salem Unit 1 volume control tank (VCT) flaw growth analysis performed to address flaws that were identified in the circumferential lower head-to-shell weld of the Salem Unit 1 VCT during refueling outage 1RF13 (1999). The LRA also states that the analyses concluded that an initial flaw would grow an insignificant amount of only 1.1×10^{-5} inches, based on 1,000 pressurization cycles. The LRA further states that the major pressurization cycles (transients) experienced by the VCT would be Inadvertent Safety Injection events and Operating Basis Earthquake cycles, and to a lesser extent, Plant Heatups and Cooldowns. The LRA concludes that the VCT flaw growth analysis will remain valid during the period of extended operation because the maximum number of pressurization cycles projected for 60 years (e.g., 312 cycles) has been demonstrated to be bounded by the 1,000 pressurization cycles limit assumed in the Salem Unit 1 VCT flaw growth analysis.

Issue:

The LRA does not identify the methodology used to perform the analyses. It is not evident from the TLAA discussion whether ASME Code, Appendix A methodology or other similar industry standard was used to perform the analyses.

LRA Table 4.3.1-3 provides the design transients and 60-year projections for NSSS Class A and Class 1 components at Salem Unit 1. The staff noted that one of the upset condition transient is reactor trip from full power, which would SCRAM the reactor from full power and cause a full depressurization of the reactor coolant system. This design transient is not considered in the projected number of pressurization cycles that were used to conclude that the VCT crack growth analyses remained valid during the period of extended operation.

Request:

(a) Clarify which methodology was used to perform the Salem 1 VCT flaw growth analysis and whether the methodology has been approved for use by the NRC. Clarify which NRC document provides the approval of methodology. If the methodology has not been approved by the NRC, justify the bases for use of the analysis methodology and the rationale for choosing an acceptance criterion of 1000 pressurization cycles.

(b) Justify why the upset condition transient of "Reactor Trip from Full Power," which would SCRAM the reactor from full power and cause a full depressurization of the reactor coolant system, was not considered in the 60-year projection of pressurization cycles that were used to conclude that the VCT crack growth analyses remained valid during the period of extended operation.

PSEG Response:

(a) Salem submitted the Unit 1 Volume Control Tank (VCT) fatigue analysis to the NRC in letter LR-N00066, dated February 28, 2000, regarding Chemical and Volume Control Tank Indications, Salem Nuclear Generating Station Unit 1 (ADAMS Accession No. ML003691659). The NRC staff did not provide a specific review and approval of the fatigue analysis.

The fatigue analysis used the 1989 edition of ASME Code, Section XI analytical technique, specifically, the net section plastic collapse approach in IWB-3640, supplemented by Appendix C of ASME Section XI to determine the allowable flaw sizes for the Salem Unit 1 VCT. The reasons for using this methodology are two-fold:

1) The tank is fabricated from Type 304 stainless steel. The VCT is classified as a Class 2 tank. Currently, there are no flaw evaluation methods for Class 2 components in ASME Section XI, and therefore, Class 1 rules in IWB-3600 are generally used for Class 2 components. In IWB-3600, there are also no specific rules for stainless steel tanks.

2) Due to the inherent ductility and toughness of stainless steels, the net section plastic collapse methodology was used for the failure criteria for stainless steel components. This methodology was chosen to determine the allowable flaw size for the tank. Although there are no specific rules for evaluating stainless steel tanks in IWB-3600, evaluation rules are available for stainless steel piping components in IWB-3640. These are based on the net section plastic collapse approach. Appendix C of ASME Section XI provides the net section plastic collapse equations for stainless steel pipes subjected to the primary membrane and bending stresses. These same equations are directly applicable to stainless steel tanks, since they are also cylindrical thin wall components.

The methodology described above and submitted to the NRC on February 28, 2000 was considered an acceptable method for evaluating the two flaws found in the Salem Unit 1 VCT since the ASME Section XI Code methodology applicable to stainless steel components was used.

Finally, the fatigue analysis submitted to the NRC on February 28, 2000 used an arbitrary value of 1000 pressurization cycles, in order to establish a bounding analysis for the operation of the plant. The crack growth even using 1000 pressurization cycles was determined to be very small at 1.1×10^{-5} inches.

(b) LRA Table 4.4.5-1, "Salem Unit 1 60-Year Projection of Transients Applicable to the Volume Control Tank Flaw Growth Analysis", lists four (4) types of transients that are considered applicable for fatigue-induced flaw growth (Inadvertent Safety Injection, Operational Basis Earthquake, Heatup, and Cooldown). The summation of the individual 60-Year projected transient cycles was 312.

During the TLAA evaluation, the upset condition transient of Reactor Trip from Full Power was initially considered, however, during further internal review, it was removed from applicability as a transient condition for the VCT. The Reactor Trip from Full Power

transient does not result in a full depressurization of the Reactor Coolant System (RCS). The Reactor Trip from Full Power transient results in RCS pressure lowering to approximately 2000 psi, and then recovering to the normal operating pressure of approximately 2250 psi with the use of pressurizer heaters and increased charging flow (lower VCT level and pressure). Since the plant design and the emergency operating procedures prevent the full depressurization of the RCS, there is no pressurization cycling of the VCT during the Reactor Trip from Full Power transient. The safety injection actuation does not occur during a normal reactor trip from full power, since RCS pressure only decreases to approximately 2000 psi. The actuation for safety injection occurs at an RCS pressure of approximately 1780 psi.

RAI 4.4.5-2

Background:

LRA Section 4.4.5 states that flaws were identified in the shell to lower head weld of the Salem Unit 1 Volume Control Tank (VCT) during 1RF13 (1999). The LRA states further that the flaws found during the inspection were subsurface and not in contact with the environment, therefore, only fatigue would be the contributing mechanism to flaw growth.

Issue:

The applicant has identified that the flaw growth analysis for the Salem 1 VCT is a TLAA for the LRA. Two of the six criteria for defining an analysis as a TLAA in 10 CFR 54.3, is that the analysis must include the effects of aging of the intended function of the component, and that the analysis must be used in a safety basis decision. Since this analysis is TLAA, presumably there should be an applicable AMR item that is specific to management of fatigue flaw growth (i.e., crack growth due to fatigue) for the flaws in the VCT.

The staff has noted that LRA Table 3.3.2-2, "Chemical and Volume", only includes an applicable line item of management of cumulative fatigue damage, and do not include any AMR items of management of crack growth due to fatigue on the Salem Unit 1 VCT.

Request:

Justify why LRA Table 3.3.2-2 does not include any AMR line item for the Salem 1 VCT in a borated treated water environment with an aging effect of crack growth due to fatigue.

PSEG Response:

In PSEG letter LR-N00066 to the NRC, dated February 28, 2000, regarding Chemical and Volume Control Tank Indications, Salem Nuclear Generating Station Unit 1 (ADAMS Accession No. ML003691659), Salem determined that the Root Cause for the Salem Unit 1 Volume Control Tank (VCT) indications (flaws) was from the welding fabrication process. Two follow-up ultrasonic examinations of the Unit 1 VCT in 2002 (1RF15) and 2008 (1RF19) found no change in embedded flaw size, and there have been no recordable indications on the Unit 2 VCT. Since the Unit 1 VCT indications were not considered service-induced or caused by fatigue, the additional aging effect of cracking (e.g., crack growth) due to fatigue or environmental conditions was not included in the LRA.

Additionally, the normal service temperature of the VCT is less than 140°F in a treated borated water environment; therefore, Salem LRA Table 3.3.2-2 does not contain a separate line item for the aging effect and mechanism of Cracking due to Stress Corrosion Cracking.