PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038

PSEG
Nuclear LLC

JUL 0 8 2010

10 CFR 50
10 CFR 51
10 CFR 54

LR-N10-0247

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 Requests for Additional Information dated June 10, 2010, June 11, 2010 and June 11, 2010 related to the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application

References:

1. 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 (TAC NOS. ME1834 AND ME1836)" dated June 10, 2010

2. 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 SECTIONS 3.1.2 AND 3.3.2 (TAC NO ME1834 / ME1836)" dated June 11, 2010

3. 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 (TAC NOS. ME1834 AND ME 1836)" dated June 11, 2010

This letter responds to the three NRC Request for Additional information packages cited in References 1, 2 and 3 above. In Reference 1, the NRC requested additional information related to Sections B.2.1.7, B.2.1.23, B.2.2.2 and B.3.1.1 of the Salem Nuclear Generating Station, Units 1 and 2 (Salem) License Renewal Application (LRA).

A141
NRR

JUL 08 2010

Salem LRA, and Reference 3 also requested additional information related to Section 3.3.2 of the Salem LRA. Enclosure A contains the responses to this request for additional information which contain updates to the Salem Nuclear Generating Station, Units 1 and 2, LRA Sections affected by the RAI responses including Appendix A and B updates.

Enclosure B provides an update to the affected portions of the License Renewal Commitment List (LRA Appendix A, Section A.5), as a result of these RAI responses.

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

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

Executed on 7/7/10

Sincerely,

Paul J. Davison

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

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

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 A

Responses to Request for Additional Information associated with the Salem Nuclear Generating Station, Units 1 and 2, License Renewal Application (LRA)

RAI B.2.1.7-01
RAI B.2.1.23-01
RAI B.2.2.2-03
RAI 3.1.1-01
RAI 3.3.2-01
RAI 3.3.2-02
RAI 3.1.2.1-01
RAI 3.3.2.2-3

RAI B.2.1.7-01

Background:

Based on a review of GALL Report Table 1, line item 80, for cast austenitic stainless steel (CASS) reactor vessel internal components, the GALL recommends use of the GALL AMP XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)" Program to manage the aging effects of loss of fracture toughness/thermal aging embrittlement. The GALL report does not have an aging effect of cracking, stress-corrosion cracking (SCC), irradiation-assisted stress-corrosion cracking (IASCC), void swelling, or changes in dimension line item with CASS as the material.

Issue:

Several line items in LRA Table 3.1.2-3, assigned generic note C, have an aging effect of cracking, SCC, IASCC, changes in dimension, and void swelling with CASS material components. These line items credit the LRA pressurized-water reactor (PWR) Vessel Internals AMP. These line items reference GALL Report NUREG-1801 Volume 2 line items which list stainless steel instead of CASS as the material. Additionally, one LRA Table 3.1.2-3 line item associated with a loss of fracture toughness/neutron irradiation embrittlement, and void swelling credits the PWR Vessel Internals Program and assigns this line item generic note E. Although there is no specific GALL line item for an aging effect of cracking, SCC, IASCC, changes in dimension and void swelling with CASS material components, the staff questions why the LRA does not credit GALL AMP XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)" Program to manage these aging effects.

Request:

Provide a justification for using the PWR Vessels Internals AMP instead of GALL AMP XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)."

PSEG Response:

Salem substituted the PWR Vessel Internals aging management program (Salem LRA Appendix B, Section B.2.1.7) for the GALL AMP XI.M13 aging management program, Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Program to manage the aging effect and mechanism of Loss of Fracture Toughness due to Thermal Aging and Neutron Irradiation Embrittlement of the CASS component types listed in Salem LRA Table 3.1.2-3, "Reactor Vessel Internals". The PWR Vessel Internals aging management program will implement the Electric Power Research Institute (EPRI) Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (MRP-227), December 2008, and the EPRI Report No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)", July 2009, which together manage the aging effects on the reactor vessel internal components. Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 7).

In accordance with MRP-227 and MRP-228, Salem will examine all component types constructed of CASS requiring aging management and compare the data with examination acceptance criteria as defined in MRP-227. Any recordable indications will be dispositioned in accordance with Salem's Corrective Action Program. Additionally, primary water chemistry will be monitored and controlled in accordance with the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2), which is consistent with the EPRI Primary Water Chemistry guidelines. Implementation of the Water Chemistry aging management program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 2).

The PWR Vessel Internals aging management program, implementing MRP-227 and MRP-228, will be adequate to address the aging effect and mechanism of Loss of Fracture Toughness due to Thermal Aging and Neutron Irradiation Embrittlement of the CASS component types within the license renewal scoping boundary of the Reactor Vessel Internals.

B.2.1.23-01

Background:

GALL AMP X1.M35, Element 4 "detection of aging effects," states that inspections of ASME Code Class 1 small-bore piping should be volumetric.

Issue:

The LRA Sections B.2.1.23 and A.2.1.23 state socket welds that fall within the weld examination sample will be examined using visual examination (VT-2). The staff notes that the proposed inspection methodology is not consistent with GALL.

Request:

Please provide technical basis how VT-2 can adequately manage aging in ASME Code Class 1 socket welds.

PSEG Response:

The aging effects of ASME Code Class 1 small-bore socket welds are adequately managed by crediting approved VT-2 visual inspections in accordance with current Risk-Informed ISI (RI-ISI) program requirements that meet the regulatory requirements of 10 CFR 50.55a, and Case N-578-1, "Cases of ASME Boiler and Pressure Vessel Code". Section 2500 of Code Case N-578-1 states that pressure testing and VT-2 visual examinations shall be performed on Class 1, 2, and 3 piping systems, and this is the method which Salem credits to manage the aging effects of ASME Code Class 1 small-bore socket welds. The VT-2 visual examinations are performed for each accessible small-bore socket weld every refuel outage during system leakage tests when the units are at operating temperature and pressure.

The operating experience at Salem Units 1 and 2 indicates that the VT-2 visual examinations has been effective in managing the ASME Code Class 1 small-bore socket welds. To date there have been no Class 1 small-bore socket weld failures on either Salem Unit 1 or 2 due to intergranular stress corrosion cracking or thermal fatigue (thermal and mechanical loading).

In addition to Salem operating experience that shows VT-2 visual examination methods adequately manage the aging of ASME Code Class 1 small-bore socket welds, industry operating experience also shows that small-bore socket welds have had no significant findings of degradation that resulted in confirmed age-related failures. To illustrate this point, an industry operating experience review was conducted by industry peers on behalf of NEI to determine the extent of failures related to Class 1 small-bore piping and the results were published in a letter dated March 2, 2010, which is documented in ADAMS Accession Number ML100680786. The intent of the review was to evaluate the position that VT-2 visual examinations conducted while the units are at operating temperature and pressure adequately manages ASME Class 1 small-bore piping. In short, the intent of the review was to verify the current code requirements for inspecting ASME Code Class 1 small-bore piping is adequate. The OE review concluded the following:

> "This review supports the position that the potential for small-bore socket weld failures does not represent a significant safety issue and surface examinations and visual inspections currently being performed are sufficient to identify and correct weld degradation prior to gross failure. The review confirms that the existing activities for managing ASME Class 1 small-bore socket welds provide reasonable assurance that the socket welds continue to support the intended functions of the systems in which they are installed."

The aforementioned OE review supports the position that VT-2 visual examinations adequately manage the aging effects of ASME Code Class 1 small-bore socket welds.

Even though station and industry OE suggest the code required VT-2 visual examinations adequately manage the aging effects in ASME Code Class 1 small-bore piping, qualified, and reliable volumetric examinations will likely be available soon to detect and characterize potential flaws in small-bore socket welds. As industry technology advances and methods become available to detect and characterize flaws in small-bore socket welds, in addition to the VT-2 visual examinations performed per the RI-ISI program requirements, Salem Units 1 and 2 will perform four volumetric examinations, two per unit, from a population of 36 susceptible Class 1 small-bore socket welds on Unit 1 and 34 susceptible Class 1 small-bore socket welds on Unit 2. Provided the technology is available, these inspections shall be performed prior to entering the period of extended operation. More specifically, the volumetric examinations will analyze Class 1 small-bore socket welds as follows:

- Two Class 1 small-bore socket welds (one per unit) for intergranular stress corrosion cracking; and
- Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)

The locations for the volumetric socket weld examinations will be determined by selecting the socket welds where the highest likelihood that small-bore socket weld degradation could exist. The location of the small-bore socket welds selected for inspection will be based upon the RI-ISI ranking criteria which considers the following; design characteristics, fabrication process, operating conditions, industry-wide service experience, results of pre-service, inservice, and augmented examinations, and their related potential degradation mechanisms.

These volumetric examinations will be evaluated for degraded conditions in accordance with the Corrective Action Program requirements. The volumetric inspection results will also be reviewed to determine if additional examinations are needed on a period frequency to ensure the Class 1 small-bore socket welds will continue to be adequately managed at Salem Units 1 and 2.

The demonstrated industry and station operating experience and compliance with the ASME Section XI Code requirements support the technical basis that VT-2 examinations adequately manage the aging effects in Class 1 small-bore socket welds at Salem Units 1 and 2. In addition, the volumetric examinations that will be performed when the technology becomes available will also provide additional assurance that the integrity, and safety functions of the Class 1 small-bore piping will continue to be adequately

maintained throughout the period of extended operation at Salem Units 1 and 2. Also, as volumetric examination requirements become endorsed in future ASME Code Section XI requirements, Salem Units 1 and 2 will modify the testing methods to comply with the applicable ASME Section XI Code requirements.

As a result of changes to the One-Time Inspection of ASME Code Class 1 Small Bore-Piping aging management program identified in this response, the second paragraph of Section A.2.1.23 of Appendix A, on page A-21 of the LRA, is revised as shown below. Revisions are indicated with bolded italics for inserted text and strikethroughs for deleted text.

A.2.1.23 One-Time Inspection of ASME Code Class 1 Small Bore-Piping

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping program is a new program that will manage the aging effect of cracking in stainless steel small-bore, less than nominal pipe size (NPS) 4 inches and greater than or equal to NPS 1 Class 1 piping through the use of a combination of volumetric examinations and visual inspections. The components include piping, piping elements and piping components. This program is a part of the Salem Risk Informed Inservice Inspection program.
The program will manage the aging effect through the identification and evaluation of cracking in small-bore Class 1 piping. The program will include one-time volumetric examination of a sample of class 1 butt welds for pipe size less than 4 inches NPS and greater than or equal to NPS 1. ***In addition, Salem Units 1 and 2 will perform four volumetric examinations, two per unit, from a population of 36 susceptible Class 1 small-bore socket welds on Unit 1 and 34 susceptible Class 1 small-bore socket welds on Unit 2. Provided the technology is available, these inspections shall be performed prior to entering the period of extended operation. More specifically, the volumetric examinations will analyze Class 1 small-bore socket welds as follows:***

- ***Two Class 1 small-bore socket welds (one per unit) for intergranular stress corrosion cracking; and***
- ***Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)***

~~In lieu of performing one-time volumetric inspections of socket welds for pipe size less than 4 inches NPS and greater than or equal to NPS 1, t~~ ***T***he ***current*** examination method and frequency ~~will be~~ , VT-2, as allowed per the code, each refueling outage as defined in Code Case N-578-1, Table 1 ***will continue to be conducted***. Any cracking identified in small-bore Class 1 piping resulting from stress corrosion or thermal and mechanical loading will result in periodic inspections. The program will effectively manage the aging effect by identifying and evaluating cracking in small-bore Class 1 piping prior to loss of intended function.

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be implemented and one-time inspections completed and evaluated prior to the period of extended operation.

As a result of changes to the One-Time Inspection of ASME Code Class 1 Small Bore-Piping aging management program identified in this response, the fourth paragraph of Section B.2.1.23 of Appendix B, on page B-114 of the LRA, is revised as shown below. Revisions are indicated with bolded italics for inserted text and strikethroughs for deleted text.

B.2.1.23 One-Time Inspection of ASME Code Class 1 Small Bore-Piping

Program Description

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping aging management program is a new program that manages the aging effect of cracking in stainless steel ASME Code Class 1 piping, piping elements and piping components less than 4 inches nominal pipe size (NPS 4) and greater than or equal to NPS 1 (Table IWB-2500-1, Examination Category B-J, Item No. B9.21) in reactor coolant and treated water environments. The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program is a new program that Salem will implement prior to the period of extended operation and within the last ten years of the current operating period. Salem has not experienced cracking of ASME Code Class 1 small-bore piping resulting from stress corrosion or thermal and mechanical loading. Should evidence of significant aging be revealed by the one-time inspection, periodic inspection will be proposed.

Since the publication of NUREG-1801 Revision 1, the interim guidance contained in EPRI Report 1000701, "Interim Thermal Fatigue Management Guideline (MRP-24)," has been supplemented by a more complete set of guidelines on thermal fatigue issues for lines connecting to the reactor coolant system (RCS); those guidelines are contained in EPRI Report 1011955, "Materials Reliability Program Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)." The program uses the more recent revision.

The Salem One-Time Inspection of ASME Code Class 1 Small-Bore Piping aging management program is included in the Risk Informed Inservice Inspection (RI-ISI) program, which is based on the EPRI RI-ISI Topical Report, EPRI TR-112657, and ASME Code Case N-578-1. Because the use of RI-ISI, in lieu of ASME Code Section XI, must be approved pursuant to 10 CFR 50.55a(a)(3), it is not considered an exception to NUREG-1801. The alternative RI-ISI program is substituted for the 1998 Edition through the 2000 Addenda of ASME Section XI Code Edition for Class 1 Examination Category B-J, Item No. B9.21 circumferential welds in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. The program includes one-time volumetric examination of a sample of Class 1 small-bore butt welds. ***In addition, Salem Units 1 and 2 will perform four volumetric examinations, two per unit, from a population of 36 susceptible Class 1 small-bore socket welds on Unit 1 and 34 susceptible Class 1 small-bore socket welds on Unit 2. Provided the technology is available, these inspections shall be performed prior to entering the period of extended operation. More specifically, the volumetric examinations will analyze Class 1 small-bore socket welds as follows:***

- ***Two Class 1 small-bore socket welds (one per unit) for intergranular stress corrosion cracking; and***

- ***Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)***

~~In lieu of performing one-time volumetric inspections of socket welds for pipe size less than 4 inches NPS and greater than or equal to NPS 1, t~~ ***T***he ***current*** examination method and frequency ~~will be~~ , VT-2, as allowed per the code, each refueling outage as defined in Code Case N-578-1, Table 1 ***will continue to be conducted***. Any cracking identified in small-bore Class 1 piping resulting from stress corrosion or thermal and mechanical loading will result in periodic inspections. This sample includes locations that are susceptible to cracking. The program includes measures to verify that unacceptable degradation is not occurring in Class 1 small-bore piping, thereby validating the effectiveness of the Water Chemistry Program, B.2.1.2, to mitigate aging-related degradation and confirming that no additional aging management programs are needed for the period of extended operation.

In addition, an Enhancement on LRA page B-114 has been added and is revised as follows:

Enhancements

~~None.~~

The One-Time Inspection of ASME Code Class 1 Small Bore-Piping is a new program that will be enhanced to include the following activity:

1. ***Salem Units 1 and 2 will perform four volumetric examinations, two per unit, from a population of 36 susceptible Class 1 small-bore socket welds on Unit 1 and 34 susceptible Class 1 small-bore socket welds on Unit 2. Provided the technology is available, these inspections shall be performed prior to entering the period of extended operation. More specifically, the volumetric examinations will analyze Class 1 small-bore socket welds as follows:***

    - ***Two Class 1 small-bore socket welds (one per unit) for intergranular stress corrosion cracking; and***
    - ***Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)***

The locations for the volumetric socket weld examinations will be determined by selecting the socket welds where the highest likelihood that small-bore socket weld degradation could exist. ***The location of the small-bore socket welds selected for inspection will be based upon the RI-ISI ranking criterion which considers the following; design characteristics, fabrication process, operating conditions, industry-wide service experience, results of pre-service, inservice, and augmented examinations, and their related potential degradation mechanisms.***

RAI B.2.2.2-03

Background:

Section A1.2.3 of NUREG-1800, Appendix A states that the "acceptance criteria" of the program and its basis should be described.

Issue:

In element 6 of the LRA AMP it states that "acceptance criteria" for loss of material are based on the original equipment design wall thickness and any corrosion allowance requirements. It is not clear to the staff what the "acceptance criteria" for determining effects of aging on aluminum components are.

Request:

Clarify what the "acceptance criteria" for determining effects of aging on aluminum components are.

PSEG Response:

This RAI is a duplicate request to RAI 2.2.2-03 provided in NRC to PSEG letter from Bennett Brady to Thomas Joyce, dated July 10, 2010 (ADAMS No. ML101440081). RAI 2.2.2-03 is addressed in PSEG letter, LR-N10-0225.

RAI 3.1.1-01

Background:

Standard Review Plan-License Renewal (SRP-LR) Section 3.1.2.2.13 identifies that cracking due to primary water stress-corrosion cracking (PWSCC) could occur in PWR components made of nickel alloy and steel with nickel alloy cladding, including reactor coolant pressure boundary components and penetrations inside the reactor coolant system (RCS) such as pressurizer heater sheathes and sleeves, nozzles, and other components. GALL Report Volume 2 Item IV.D1-06 recommends Chapter XI.M2, "Water Chemistry" for PWR primary water for managing the aging effect of cracking in the nickel alloy steam generator (SG) divider plate exposed to reactor coolant.

In LRA Table 3.1.1, Item 81, the applicant credits its Water Chemistry Program to manage cracking due to stress corrosion cracking in nickel alloy steam generator (SG) primary channel head divider plate exposed to reactor coolant in the steam generators.

In UFSAR Section 5.5.2.2.1, the applicant describes that the divider plate is made with Inconel 690 for Unit 2 replacement steam generators. The staff notes that the use of this nickel alloy 690 should prevent the aging effect of PWSCC.

However, there is no information about the construction material of the divider plate for Unit 1 steam generators.

Issue:

From recent operating experience in steam generators with a similar design to that of Salem Unit 1 steam generators, extensive cracking due to PWSCC has been identified in SG divider plates made with nickel alloy 600, even with proper primary water chemistry. Specifically, cracks have been detected in the stub runner, very close to the tubesheet/stub runner weld and with depths of almost a third of the divider plate thickness. Therefore, the staff notes that the Water Chemistry Program alone may not be effective in managing the aging effects of cracking due to PWSCC in SG divider plates.

Such cracks could impact adjacent items, such as the tubesheet and the channel head, if they propagate to the boundary with these items. For the tubesheet, PWSCC cracks in the divider plate could propagate to the tubesheet cladding with possible consequences to the tube/tubesheet welds. For the channel head, the PWSCC cracks in the divider plate could propagate to the SG triple point and potentially affect the pressure boundary of the SG channel head.

Request:

Please discuss the materials of construction of your Unit 1 SG divider plate assembly. If these materials are susceptible to cracking (e.g., nickel alloy 600 or the associated nickel alloy weld materials), please discuss the potential for cracking in the divider plate to propagate into other components (e.g., tubesheet cladding).

If propagation into these other components cannot be ruled out, please describe an inspection program (examination technique and frequency) to ensure that there are no cracks propagating into other items (e.g., tubesheet and channel head) that could challenge the integrity of other adjacent items.

PSEG Response:

The Salem Unit 1 Steam Generators (SG) are Westinghouse Model F with a divider plate installed in the hemispherical channel head located on the bottom of the SG. The channel head is constructed of a cast low alloy steel with austenitic stainless steel cladding. The channel head is partitioned by the divider plate and stub runner. The divider plate and stub runner components are constructed of Alloy 600. The welds are 82/182 that attach the divider plate and stub runner to each other, and to the channel head and tubesheet.

Salem LRA Table 2.3.1-4, "Steam Generators", lists the intended function for the Steam Generators (Primary Channel Head Divider Plate) as Direct Flow. The divider plate for the Salem Steam Generators does not serve an Intended Function of Pressure Boundary.

As identified in LRA Table 3.2.1-4, "Steam Generators", the Component Type Steam Generators (Primary Channel Head Divider Plate) is fabricated from nickel alloy (Alloy 600). The aging management review determined that the nickel alloy material of this component is potentially susceptible to Cracking/Stress Corrosion Cracking (SCC) and Loss of Material/Pitting and Crevice Corrosion in the Reactor Coolant environment. Both of these aging effects are managed by the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2), as indicated in LRA Table 3.2.1-4, consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Volume II, Items IV.D1-06 and IV.C2-15.

Westinghouse has recently evaluated the Alloy 600 divider plate cracking phenomena observed in the foreign operating experience, reviewed the materials of construction for susceptible steam generators, including the Salem Unit 1 Westinghouse Model F steam generators, and evaluated crack growth scenarios in the divider plate region. Westinghouse concluded that currently available data on the divider plate cracking phenomena in the foreign fleet suggest that the postulated concern is not likely. It is not feasible for the cracks to grow in such a fashion as to propagate through the divider plate material and adjoining welds to the channel head and tubesheet structures because of the expected loading on the structure. First principles examination of the potential divider plate cracking phenomena indicates that there should be no impact during the period of extended operation.

Westinghouse's qualitative evaluation discusses that crack propagation into the channel head and tubesheet reactor coolant pressure boundary regions is not expected in domestic steam generators based on:

- Crack propagation in the direction of base materials or cladding would require the crack to grow contrary to the influence of the primary stress field, which is not consistent with fracture mechanics principles.

- Based on service experience, stress corrosion cracks stop when they meet a non-susceptible material (i.e., for Salem Unit 1, austenitic stainless steel cladding on channel head or the low-alloy steel base (structure) materials of the nickel alloy cladded tubesheet).

- Existing analysis data suggested in the EPRI studies of divider plate cracking that the stresses in the divider plate materials become compressive within a short distance of the face of the divider plate if a crack were to propagate into the thickness of the weld region. Similarly, the stress field in the triple point connection of the channel head, divider plate, tubesheet, and stub runner is also compressive, therefore, this data concludes that a postulated crack in a PWSCC-sensitized divider plate material will remain in its original plane.

Therefore, the potential for cracking of the Salem Unit 1 Model F Steam Generator divider plate propagating into adjoining components and resulting in loss of the integrity of the reactor coolant pressure boundary would not be expected to occur, and therefore the Steam Generators (Primary Channel Head Divider Plate) does not require an aging management program consisting of inspections for crack propagation.

RAI 3.3.2-01

Background:

Salem Nuclear Generating Station (SNGS) license renewal application (LRA) Tables 3.3.2-1 (Auxiliary Building Ventilation), 3.3.2-7 (Containment Ventilation), 3.3.2-15 (Fuel Handling Ventilation), 3.3.2-26 (Switchgear and Penetration Area Ventilation) contain aging management review (AMR) lines for elastomer door seals and flexible connections exposed to air with borated water leakage. For this combination of component, material, and environment, the LRA states that there is no aging effect and no aging management program (AMP) is required. These same SNGS LRA tables contain elastomer door seals and flexible connections exposed to air/gas – wetted (internal). For this combination of component, material, and environment, the LRA credits the Periodic Inspection Program to manage hardening and loss of strength due to elastomer degradation.

Issue:

The LRA does not provide a technical basis for concluding there is no aging effect for elastomer door seals and flexible connections exposed to air with borated water leakage. Also, it is not clear to the staff whether the elastomer door seals and flexible conduit determined to have no aging effect when exposed to air with borated water leakage are the same components as those exposed to air/gas – wetted internals.

Request:

a) Provide the technical basis for concluding that elastomer door seals and flexible conduit have no aging effect when exposed to air with borated water leakage.

b) Clarify whether the components exposed to air with borated water leakage are the same as those exposed to air/gas – wetted (internal). If they are the same, explain why one environment results in an aging effect and the other does not.

PSEG Response:

a) The following Summary of Aging Management Evaluation tables list elastomer door seals and flexible connections exposed to an Air with Borated Water Leakage external environment as component types requiring aging management:

- Table 3.3.2-1, Auxiliary Building Ventilation System
- Table 3.3.2-7, Containment Ventilation System
- Table 3.3.2-15, Fuel Handling Ventilation System
- Table 3.3.2-26, Switchgear and Penetration Area Ventilation System

The line items for the elastomer door seals and flexible connections with an external environment of Air with Borated Water Leakage are assigned no aging effects requiring management in the Salem LRA. These component types also list an external environment of Air-Indoor and an internal environment of Air/Gas – Wetted. The elastomer component types are assigned an aging effect of hardening and loss of strength for the Air-Indoor external environment and the Air/Gas – Wetted internal environment.

For all components in the Auxiliary Building (including the Inner Penetration Area), Containment Structure, and Fuel Handling Building, two external environments are applied: Air-Indoor and Air with Borated Water Leakage. The Air with Borated Water Leakage is included specifically to cover metallic component types whose external surfaces are susceptible to boric acid wastage as defined in the EPRI Mechanical Tools "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools", Revision 4, Appendix E, Section 3.1.10, "Boric Acid Wastage". These component types would then be monitored for loss of material under the Boric Acid Corrosion aging management program (Salem LRA Appendix B, Section B.2.1.4). Since elastomers are non-metallic there are no aging effects listed in the LRA.

Although the line items for the elastomer components indicate that there are no aging effects for the door seals and the flexible connections in external environment of Air with Borated Water Leakage, the intent is to indicate that there are no additional aging effects introduced by this environment beyond the one listed under the Air-Indoor environment. This is described in the plant specific note (e.g., G,3) for each of the line items with an external environment of Air with Borated Water Leakage. NUREG-1801 does not include this material and environment combination. A review of industry operating experience , including Salem, does not indicate boric acid corrosion as a potential aging effect requiring management for elastomer components.

Therefore, boric acid corrosion is not included as an aging effect requiring management for elastomer components. The aging effects of hardening and loss of strength are addressed under the Air-Indoor environment and the use of the Periodic Inspection aging management program (Salem LRA Appendix B, Section B.2.2.2) is adequate for detection of any degradation in the door seals and flexible connections prior to the loss of intended function.

b) The elastomer door seal and flexible connection components exposed to an Air with Borated Water Leakage external environment are the same as the components exposed to an Air-Indoor external environment and an Air/Gas-Wetted internal environment. Salem license renewal project methodology dictates that an internal and external environment is identified for all component types with the exception of bolting. All components in the Auxiliary Building (including the Inner Penetration Area), Containment Structure, and Fuel Handling Building are assigned two external environments: Air-Indoor and Air with Borated Water Leakage. The Air with Borated Water external environment is listed specifically to address the potential for boric acid corrosion. All other aging effects applicable to the external surface of the subject components are addressed under the Air-Indoor external environment.

Air with Borated Water Leakage is not identified as an environment for this material (elastomer) in NUREG-1801, and therefore, the G Note is appropriate for the combination of the material and environment. For the Air-Indoor (external) environment, the door seals and flexible connections are aligned to NUREG-1801 VII.F2-7, Item Number 3.3.1-11. The Periodic Inspection aging management program is used to manage the aging effects of hardening and loss of strength for these items. The plant specific note (e.g., G,3) for the Air with Borated Water Leakage external environment line item states that there are no additional aging effects on these elastomer component types. The Periodic Inspection aging

management program is used to inspect the exterior and interior of these component types for aging effects. Therefore, any degradation of the door seals or flexible connections will be detected and corrected prior to loss of component intended function.

RAI 3.3.2-02

Background:

SNGS LRA Table 3.3.2-26 (Switchgear and Penetration Area Ventilation) contains AMR lines for elastomer door seals and flexible connections exposed to air with borated water leakage and to air/gas – wetted.

Issue:

The LRA does not provide sufficient information for the staff to evaluate the effect of these environments for components in this system.

Request:

Explain the basis for including these environments for elastomer door seals and flexible connections in the Switchgear and Penetration Area Ventilation System, in particular, how the elastomers are exposed to the borated water leakage and wetted air/gas mixture.

PSEG Response:

The basis for including the Air with Borated Water Leakage environment for elastomer door seals and flexible connections in the Switchgear and Penetration Area Ventilation System is due to the location of the components. All components in the Auxiliary Building (including the Inner Penetration Area), Containment Structure, and Fuel Handling Building are assigned two external environments: Air-Indoor and Air with Borated Water Leakage. While the probability of the elastomer components in this system being exposed to borated water due to leakage is extremely small, the environment has been included for completeness. The Air with Borated Water Leakage environment is assigned specifically to address potential for boric acid corrosion. All other aging effects are addressed under the Air-Indoor environment.

The ventilation components are assigned an internal environment of Air/Gas-Wetted in accordance with the Salem license renewal project methodology. All internal air environments are classified as Air/Gas – Wetted unless the air is processed through filters and driers to remove moisture and contaminants and is maintained to high quality.

The elastomer components are not exposed to an Air with Borated Water Leakage and Air/Gas-Water mixture. The internal surfaces of the components are exposed to the Air/Gas-Wetted environment. The external surfaces of the components are exposed to the indoor air environment of a building containing borated water systems. Therefore, the license renewal environment of Air with Borated Water Leakage is included for completeness and to address the potential for boric acid corrosion.

RAI 3.1.2.1-01

Background:

License renewal application (LRA) Table 3.1.1, item 3.1.1-80, addresses loss of fracture toughness due to thermal aging and neutron embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals exposed to reactor coolant and neutron flux. LRA item 3.1.1-80 also indicates that the PWR Vessel Internals Program (B.2.1.7) will be substituted to manage the aging effect in CASS reactor vessel internal components exposed to reactor coolant and neutron flux. LRA Table 3.1.2-3 further addresses one aging management review (AMR) line item to manage the loss of fracture toughness due to thermal aging and neutron embrittlement of CASS incore guide cruciforms exposed to reactor coolant and neutron flux.

Issue:

The staff noted that the LRA Table 3.1.2-3 does not address loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of rod control cluster assembly (RCCA) guide tube assemblies (lower flanges), upper internals assembly (upper support column bases), and upper internals assembly (static flow mixers) although the LRA table indicates that the materials of the components is CASS and the PWR Vessel Internals Program is credited to manage the changes in dimensions due to void swelling and cracking due to stress corrosion cracking (SCC) and irradiation-assisted stress corrosion cracking.

The staff also noted that the aging effects addressed for the components in the LRA table indicate that neutron irradiation is applicable to the components because the void swelling and irradiation-assisted stress corrosion cracking require the exposure of the components to neutron irradiation. In addition, the staff noted that the omission of the components in managing loss of fracture toughness for the CASS reactor vessel internals is not consistent with GALL Vol. 1, Table 1 ID 80, that addresses loss of fracture toughness due to thermal aging and neutron embrittlement for CASS reactor internals including upper internals assembly, lower internal assemblies and control rod guide tube assembly.

Request:

1. Describe what program(s) is used to manage loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS rod control cluster assembly (RCCA) guide tube assemblies (lower flanges), upper internals assembly (upper support column bases), and upper internals assembly (static flow mixers) that are described in LRA Table 3.1.2-3. In addition, if the applicant does not have a program to manage loss of fracture toughness for the CASS vessel internal components, justify why it is not required to manage loss of fracture toughness for the components.

PSEG Response:

In Salem LRA Table 3.1.2-3, "Reactor Vessel Internals", the only listed component type constructed of Cast Austenitic Stainless Steel (CASS) subject to Reactor Coolant and Neutron Flux is the Reactor Vessel Internals (incore guide cruciforms). The associated aging effect and mechanism is Loss of Fracture Toughness due to Thermal Aging and Neutron Irradiation Embrittlement.

The following three (3) component types constructed of CASS were inadvertently omitted from LRA Table 3.1.2-3; the RCCA guide tube assemblies (lower flanges), the Upper Internals Assembly (static flow mixers), and the Upper Internals Assembly (upper support column bases). Therefore, LRA Table 3.1.2-3 is revised to add these three component types constructed of CASS within the Reactor Vessel Internals with the additional environment of Reactor Coolant and Neutron Flux along with the associated aging effect and mechanism of Loss of Fracture Toughness due to Thermal Aging and Neutron Irradiation Embrittlement.

Salem will substitute the PWR Vessel Internals aging management program (Salem LRA Appendix B, Section B.2.1.7) in lieu of the GALL AMP XI.M13, Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Program to manage the aging effect and mechanism of Loss of Fracture Toughness due to Thermal Aging and Neutron Irradiation Embrittlement for the four (4) CASS component types listed in the updated Salem LRA Table 3.1.2-3. The PWR Vessel Internals aging management program will implement the Electric Power Research Institute (EPRI) Report No. 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (MRP-227), December 2008, and the EPRI Report No. 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)", July 2009, which together manage the aging effects and mechanisms of the reactor vessel internal CASS components subject to the environment of Reactor Coolant and Neutron Flux. Implementation of this program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 7).

In accordance with MRP-227 and MRP-228, Salem will examine all component types constructed of CASS requiring aging management and compare the data with examination acceptance criteria as defined in MRP-227. Any recordable indications will be dispositioned in accordance with Salem's Corrective Action Program. Additionally, primary water chemistry will be monitored and controlled in accordance with the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2), which is consistent with the EPRI Primary Water Chemistry guidelines. Implementation of the Water Chemistry aging management program is a commitment in the Salem LRA, Appendix A, Section A.5 (Item 2).

The revision to Salem LRA Table 3.1.2-3 to reflect the three (3) additional CASS component types in an environment of Reactor Coolant and Neutron Flux is as follows:

Table 3.1.2-3 Reactor Vessel Internals

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RCCA guide tube assemblies (lower flanges)	*Structural Support to maintain core configuration and flow distribution*	*Cast Austenitic Stainless Steel (CASS)*	*Reactor Coolant and Neutron Flux*	*Loss of Fracture Toughness/Thermal Aging and Neutron Irradiation Embrittlement*	*PWR Vessel Internals*	*IV.B2-21*	*3.1.1-80*	*E, 5*
Upper Internals Assembly (static flow mixers)	*Structural Support to maintain core configuration and flow distribution*	*Cast Austenitic Stainless Steel (CASS)*	*Reactor Coolant and Neutron Flux*	*Loss of Fracture Toughness/Thermal Aging and Neutron Irradiation Embrittlement*	*PWR Vessel Internals*	*IV.B2-21*	*3.1.1-80*	*E, 5*
Upper Internals Assembly (upper support column bases)	*Structural Support to maintain core configuration and flow distribution*	*Cast Austenitic Stainless Steel (CASS)*	*Reactor Coolant and Neutron Flux*	*Loss of Fracture Toughness/Thermal Aging and Neutron Irradiation Embrittlement*	*PWR Vessel Internals*	*IV.B2-21*	*3.1.1-80*	*E, 5*

RAI 3.3.2.2-3

Background:

LRA Section 3.3.2.2.3.2 addresses cracking due to SCC that could occur in stainless steel and stainless clad steel heat exchanger components exposed to treated water greater than 60°C (>140°F). The LRA states that Item Number 3.3.1-5 is not applicable to the Auxiliary Systems for stainless steel and stainless clad steel heat exchanger components exposed to treated water because this component, material, environment, and aging effect/mechanism for Auxiliary System components are managed within Item 3.3.1-90 and uses the Water Chemistry Program to manage the aging effects of components exposed to treated borated water at temperatures greater than 60° C (> 140° F). The staff also noted the LRA states that the Water Chemistry aging management program (AMP) "does not provide for detection of aging effects. However, components located in selected areas at Salem will receive a one-time visual inspection...."

The staff reviewed LRA Section 3.3.2.2.3.2 against the criteria in Standard Review Plan-License Renewal (SRP-LR) Section 3.3.2.2.3.2 which states that cracking due to SCC could occur in stainless steel and stainless clad steel heat exchanger components exposed to treated water greater than 60°C (>140°F). The SRP-LR further states that the GALL Report recommends further evaluation of a plant-specific AMP to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR).

Issue:

The staff noted the applicant relies on Water Chemistry Program alone to managed cracking due to SCC in stainless steel and stainless clad steel heat exchanger components exposed to treated water greater than 60°C (>140°F) whereas a plant-specific program in accordance with SRP-LR, Appendix A.1 has a "detection of aging effects" element which is not addressed in the Water Chemistry Program.

Request:

Provide the method that will be used to detect cracking or provide justification for not performing activities that will detect cracking SCC in stainless steel and stainless clad steel heat exchanger components exposed to treated water greater than 60°C (>140°F).

PSEG Response:

The stainless steel or stainless clad steel heat exchanger components exposed to treated water greater than 60°C (140°F) in the Auxiliary Systems that align with LRA Table 3.3.1, Item Number 3.3.1-5, have been evaluated with the Steam and Power Conversion System grouping. The further evaluation in section 3.3.2.2.3.2 of the LRA incorrectly references Item Number 3.3.1-90, which is used for components exposed to treated borated water greater than 60°C (140°F). The Sampling and Demineralized Water Systems, which contain stainless steel heat exchanger components exposed to treated water greater than 60°C (140°F) in the Auxiliary Systems, are correctly aligned with LRA Table 3.4.1, Item Number 3.4.1-14 and associated section 3.4.2.2.6. The programs that will be used to detect cracking for these components are the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2), and the One-Time Inspection aging management program (Salem LRA Appendix B, Section

B.2.1.20). In order to address the incorrect Item Number reference, Section 3.3.2.2.3.2 on page 3.3-37 of the LRA will be revised as follows:

Revisions are indicated with bolded italics for inserted text and strikethroughs for deleted text.

Item Number 3.3.1-5 is not applicable to the Auxiliary Systems for stainless steel and stainless clad steel heat exchanger components exposed to treated water. ~~This component, material, environment, and aging effect/mechanism for Auxiliary System components are managed within Item Number 3.3.1-90 and uses the Water Chemistry program to manage the aging effects of components exposed to treated borated water at temperatures >60°C (140°F) .~~ ***There are no stainless steel heat exchanger components exposed to treated water greater than 60°C (140°F) evaluated in the Auxiliary Systems. The stainless steel heat exchangers components exposed to treated water greater than 60°C (140°F) and subject to stress corrosion cracking in the Sampling System and Demineralized Water Systems have been evaluated with the Steam and Power Conversions System grouping. See Table 3.4.1, Item Number 3.4.1-14, which utilizes the Water Chemistry aging management program and One-Time Inspection aging management program to manage the aging effects of these components. The Water Chemistry program and One-Time Inspection programs are described in Appendix B.***

Enclosure B

Salem Nuclear Generating Station, Units 1 and 2, License Renewal Commitment List Changes

This Enclosure includes changes to the Salem Nuclear Generating Station, Units 1 and 2, License Renewal Application (LRA) Appendix A, Section A.5 License Renewal Commitment List, as a result of the PSEG response to the following RAI:

RAI B.2.1.23-01

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.

As a result of the response to **RAI B.2.1.23-01** provided in Enclosure A of this letter, LRA Table A.5 License Renewal Commitment List, line Item 23, is revised as shown below. The specific RAI that led to the commitment modifications are listed in the "SOURCE" column adjacent to the beginning of the new text. Any other actions described in this submittal represent intended or planned actions. They are described for the NRC's information and are not regulatory commitments.

A.5 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA APP. A)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
23	One-Time Inspection of ASME Code Class 1 Small Bore-Piping	The program will manage the aging effect through the identification and evaluation of cracking in small-bore Class 1 piping. The program will include one-time volumetric examination of a sample of class 1 butt welds for pipe size less than 4 inches NPS and greater than or equal to NPS 1. ***The One-Time Inspection of ASME Code Class 1 Small Bore-Piping is a new program that will be enhanced to include the following activity:*** **Salem Units 1 and 2 will perform four volumetric examinations, two per unit, from a population of 36 susceptible Class 1 small-bore socket welds on Unit 1 and 34 susceptible Class 1 small-bore socket welds on Unit 2. Provided the technology is available, these inspections shall be performed prior to entering the period of extended operation. More specifically, the volumetric examinations will analyze Class 1 small-bore socket welds as follows:** • **Two Class 1 small-bore socket welds (one per unit) for intergranular stress corrosion cracking; and** • **Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)**	A.2.1.23	The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be implemented and one-time inspections completed and evaluated prior to the period of extended operation. ***Program to be enhanced prior to the period of extended operation.*** ***The inspection schedule will be consistent with the Salem ISI Program requirements.***	Section B.2.1.23 ***Salem letter LR-N10-0247 RAI B.2.1.23-01***