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

Salem Nuclear Generating Station Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

SUBJECT: Response to Salem Nuclear Generating Station, Unit Nos. 1 and 2 – Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals (TAC Nos. MF5149 and MF5150)

REFERENCES:

1. Salem Nuclear Generating Station, Unit Nos. 1 and 2 – Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals (TAC Nos. MF5149 and MF5150), dated March 31, 2015

2. LR-N14-0183, Submittal of PWR Vessel Internals Inspection Plans for Aging Management of Reactor Internals at Salem Generating Station, Units 1 and 2, dated August 11, 2014

On March 31, 2015, the Nuclear Regulatory Commission (NRC) provided to Mr. Thomas Joyce of PSEG Nuclear LLC (PSEG) a request for additional information (Reference 1). PSEG hereby formally documents its response to the request for additional information. Attachment 1 contains the NRC's questions followed by PSEG's response.

There are no regulatory commitments contained in this letter.

Should you have any questions regarding this submittal, please contact Mr. D. Lafleur at (856) 339-1754.

Sincerely,

John F. Perry

John F. Perry
Site Vice President – Salem
Attachments (1)

cc Mr. D. Dorman, Administrator – Region 1, NRC
Ms. C. Parker, Licensing Project Manager – Salem, NRC
Mr. P. Finney, USNRC Senior Resident Inspector, Salem (X24)
Mr. P. Mulligan, Manager IV, NJBNE
Mr. R. Braun, President and Chief Nuclear Officer – Nuclear
Mr. T. Cachaza, Salem Commitment Tracking Coordinator
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. D. Lafleur, Salem Regulatory Assurance

Attachment 1

Response to Salem Nuclear Generating Station, Units 1 and 2 - Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals (TAC Nos. MF5149 and MF5150)

RAI-1

Historically, the following materials used in the pressurized-water reactor (PWR) reactor vessel internal (RVI) components were known to be susceptible to some of the aging degradation mechanisms that are identified in the MRP-227-A topical report. In this context, the NRC staff requests that the licensee provide a list of any additional RVI components (not listed in MRP-227-A and MRP-191, Revision 0, "Material Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design" (ADAMS Accession No. ML091910130)), which are manufactured from the following materials. If any of these materials are identified as an additional RVI component at Salem, Unit 1 or 2, please provide information on the type of aging effect that was detected, and the type of AMP implemented on these components.

a) Nickel base alloys—Inconel 600; Weld Metals—Alloy 82 and 182 and Alloy X-750

b) Stainless steel type 347 material (excluding baffle-former bolts)

c) Precipitation hardened (PH) stainless steel materials—17-4 and 15-5

d) Type 431 stainless steel material

e) Alloy A-286, ASTM A 453 Grade 660, Condition A or B

Response to RAI-1

In support of Applicant/Licensee Action Item (A/LAI) 2 of MRP-227-A (Reference 1), the Salem Units 1 and 2 RVI components and their materials of fabrication were reviewed. In the comparison of the Salem Units 1 and 2 RVI components, with the components and materials of fabrication listed in MRP-191, Table 4-4 (Reference 2), no additional Salem Units 1 and 2 RVI components were identified during that review as manufactured from the materials listed in RAI-1.

RAI-2

The NRC staff requests the licensee to provide a list of RVI components at Salem, Units 1 and 2 that have been inspected, under the American Society of Mechanical Engineers Code, Section XI In-service Inspection (ISI) program, thus far, and the inspection results. Please include any RVI component categorized under the "Existing" inspection category in the MRP-227-A topical report.

Response to RAI-2

The RVI components categorized under the "Existing" inspection category in Table 4-9 of MRP-227-A topical report that are applicable to Salem Units 1 and 2 and are included as part of the ASME Section XI program, with exception of the Flux Thimble Tubes, are listed below. The Flux Thimble Tube Inspection Program is a new aging management program and will be implemented prior to the period of extended operation, NRC Safety Evaluation Report (ML110900295), section 3.0.3.1.14.

- Core barrel flange
- Upper support ring or skirt
- Lower core plate

- Core barrel support lugs (includes clevis insert bolts)
- Upper core plate alignment pins

Each of these MRP-227-A "Existing Programs" items have been visually examined (VT-3) under the ASME Section XI In-service Inspection Program at Salem Units 1 and 2. The examinations are performed on all accessible surfaces of these items compliant with In-service Inspection 10-year interval requirements. The most recent third 10-year interval examinations, with the RV core barrel removed, for both Salem units were extended in accordance with relief request SC-I3R-95 and NRC Safety Evaluation Report (ML100491550). The Section XI In-service Inspection Program Plans that includes the list of all RVI components that have been examined thus far as well as the examination results are contained in the plant records. A review of all inspections completed to date concludes there have been no relevant indications identified for the Salem Units 1 and 2 components identified as “Existing” in Table 4-9 of MRP-227-A during the periodic or interval examinations.

RAI-3

According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," (ADAMS Accession No. ML061880278) susceptibility to stress corrosion cracking (SCC) in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment (HTH) processes that are used on Alloy X-750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to its Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Additionally, Appendix A of topical report MRP-227-A, which summarizes the operational experience due to age-related degradation mechanisms, addressed that Alloy X-750 used for the clevis insert bolt assembly in one unit failed due to primary water stress- corrosion cracking (PWSCC).

Therefore, the NRC staff requests that the licensee provide information related to the type of heat treatment process that was used for the Alloy X-750 clevis insert bolting at Salem, Units 1 and 2. If the existing clevis insert bolts at Salem, Units 1 and 2 did not undergo an HTH process, please indicate if these bolts have been inspected, or if there are plans to inspect these bolts, (in addition to the inspections for monitoring aging due to wear) for degradation due to PWSCC.

Response to RAI-3

A review of the drawings and specifications for the Salem Units 1 and 2 clevis insert bolts indicates that these Alloy X-750 bolts did not receive an HTH process heat treatment. Rather, consistent with other Alloy X-750 components from the same time period, they received one of the lower temperature heat treatments used prior to the application of HTH processing in nuclear power plant components.

The potential aging degradation of the clevis insert bolts at Salem Units 1 and 2 will be managed according to the NRC-approved topical report, MRP-227-A. The clevis insert bolts are listed in Table 4-9 of MRP-227-A as an Existing Programs component. This component assembly is already included under the American Society of Mechanical Engineers (ASME) Section XI program at Salem Units 1 and 2. As noted in Table 4-9, the bolt was screened in for stress relaxation and associated cracking, but the VT-3 inspection is intended to detect wear.

The inclusion of the clevis insert bolts in MRP-227-A was based on the screening and expert panel evaluation documented in MRP-191 (Reference 2), which determined that the bolts could be susceptible to SCC and wear. It was determined that the current ASME Section XI inspection was an adequate existing program to monitor the bolts. Additional guidance for this existing inspection was provided in Westinghouse technical bulletin TB-14-5 (Reference 3) after the experience at the one plant mentioned in MRP-227-A, Appendix A and the subsequent causal investigation. This guidance did not change the type or timing of the inspection, but gave recommendations for the scope and focus of the examination in order to detect known indications of failure. PSEG Nuclear LLC (PSEG) has enhanced the in-service inspection (ISI) program in accordance with guidance from TB-14-5 (Reference 3).

The clevis insert assembly has been inspected as part of the ASME Section XI 10-year ISI in 1RFO14 (Spring 2001) for Salem Unit 1 and 2RFO12 (Spring 2002) for Salem Unit 2 using a VT-3 inspection. This inspection is capable of detecting misaligned bolt heads, cracked lock bars, and missing lock bars and bolt heads, which would occur if stress relaxation and cracking had occurred. None of these inspections have returned relevant indications for the clevis insert bolts. Future inspections of the clevis insert assembly will include recommendations provided in Westinghouse technical bulletin TB-14-5 (Reference 3), and are scheduled to occur in 1RFO26 (Spring 2019) for Salem Unit 1 and 2RFO25 (Fall 2021) for Salem Unit 2.

RAI-4

In MRP-2013-025, “MRP-227-A Applicability Template Guideline,” (ADAMS Accession Number ML13322A454) report MRP has identified two generic questions that all Combustion Engineering, Inc. and Westinghouse design plants referencing topical report MRP-227-A must address to close AI 1 related to plant-specific applicability of the topical report. If the answer to either or both questions is yes, then further evaluation will be necessary to demonstrate the applicability of MRP-227-A to Salem, Units 1 and 2. The NRC staff therefore requests the following information:

1. Do the Salem, Units 1 and 2, RVI components have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 ksi? In particular, the plant-specific information on the extent of cold work on its RVI components. The licensee can apply “Option 1” or “Option 2,” as addressed in Appendix A of the MRP-2013-025 report. If “Option 2” is applicable to Salem, Units 1 and 2, the licensee should list plant-specific RVI components that have been exposed to cold work equal to or greater than 20 percent. Plant-specific information related to this issue as addressed in “Option 2” in Appendix A, should be provided.
2. Have Salem, Units 1 and 2, ever utilized atypical design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant, including power changes/uprates? The following guidelines provided by MRP should be followed. The licensee is requested to use the MRP document dated October 14, 2013, MRP-2013-025, and it can apply “Option 1” or “Option 2,” as addressed in Appendix B of the MRP-2013-025 report.

Option 1:

Salem, Units 1 and 2, comply with the MRP-227-A assumptions regarding core loading/core design. Neutron fluence and heat generation rates are concluded to be Option A or Option B.

Option A: acceptable based on the following assessment to the limiting MRP guidance threshold values.

Option B: unacceptable based on an assessment to the limiting MRP guidance threshold values.

If Option A as addressed under "Option 1" is applicable, the following plant-specific values should be submitted: (a) active fuel to fuel alignment plate distance; (b) average core power density; and, (c) heat generation figure of merit.

If Option B under "Option 1" is applicable to Salem, Units 1 and 2, the licensee should justify the usage of its fuel management program.

Option 2:

Salem, Units 1 and 2 do not comply with the MRP-227-A assumptions regarding core loading/core design. The licensee should provide a technical justification for the application of MRP-227-A criterion to Salem, Units 1 and 2.

Response to RAI-4

PSEG, as part of the Pressurized Water Reactor Owner's Group (PWROG) Materials Subcommittee (MSC), has authorized Westinghouse to develop a report to provide a basis for a response to the NRC's request in RAI-4. The project is under Project Authorization PA-MSC-0983, Revision 1. PSEG will formally respond to RAI-4 when the PWROG project is completed. A formal response is expected late in 2015 or early 2016.

RAI-5

In response to AI 3, the licensee indicated that the control rod guide tube (CRGT) support pins that were fabricated from Alloy X-750 material were replaced by strain-hardened 316 stainless steel (SS) material. The licensee also indicated that Salem, Units 1 and 2, are consistent with the requirements in Table 4-9 of topical report MRP-227-A. Table 4-9 of topical report MRP-227-A, however, does not include CRGT support pins. As indicated in Section 3.2.5.3 of the NRC staff's safety evaluation of topical report MRP- 227-A, licensees shall evaluate the adequacy of their existing programs to manage aging degradation during the period of extended operation (PEO) for both Alloy X-750 and type 316 SS support pins.

Please provide the plant specific evaluation of the existing program under which the replacement CRGT support pins are currently inspected. In addition, please provide an evaluation of the adequacy of the existing inspection program to ensure that the aging degradation is adequately managed during the PEO for the 316 SS CRGT support pins.

Response to RAI-5

The MRP-227-A, subsection 4.4.3 (Reference 1) guidance for guide tube support pins (split pins) in Westinghouse plants is limited to plant-specific recommendations. The owner is directed to review and follow the original equipment manufacturer (OEM) recommendations for aging management and subsequent performance monitoring. Results of the detailed categorization and ranking of internals components contained in MRP-227-A, Table 3-3 (Reference 1) identifies only X-750 split pins as requiring specific actions to manage material

aging in the PEO; thus, no inspection or monitoring of the 316 SS variant for CRGT support pins is included in MRP-227-A, Table 4-9. Salem Units 1 and 2 followed the OEM recommendation to replace the originally installed X-750 CRGT support pins with support pins fabricated from strain-hardened 316 SS material (References 4, 5, 6 and 7).

As listed in MRP-191, Table 5-1 (Reference 2), the 316 SS guide tube support pins were screened in for the aging degradation mechanisms of wear, fatigue and irradiation stress relaxation/irradiation creep (ISR/IC). In MRP-232, the 316 SS CRGT support pins were categorized as MRP-191 Category A, “no additional measures.” The OEM recommendations do not require subsequent inspection of the 316 SS support pins. Long-term material behavior has been extensively studied, utilizing past testing and field experience; all identified degradation mechanisms, including irradiation-assisted stress corrosion cracking (IASCC), SCC, wear, fatigue, ISR/IC and embrittlement, have been assessed. It was concluded that the 316 SS CRGT support pins will perform all intended functions for the designated Salem Unit 1 and 2 PEO with no requirement for post-installation inspections (Reference 8). Salem existing programs comply with the OEM recommendations and MRP-227-A adequacy evaluation requirements for aging management of reactor internals CRGT 316 SS support pins.

RAI-6

In response to AI 5, the licensee referenced a Westinghouse letter with proposed acceptance criteria for 304 SS hold down springs. Please provide an explanation of the methodology for developing the acceptance criteria for the measurement of loss of compressibility of the 304 SS hold down springs.

Response to RAI-6

The general approach of WCAP-17096-NP (Reference 9) is followed in determining the acceptance criteria of the hold down spring. The decrease in hold down spring (HDS) height is assumed to occur linearly over time. The approach used to develop the HDS height acceptance criteria is to consider the actual HDS height at plant start-up and the required HDS height at the end of 60 years. A linear interpolation at the time of the HDS height measurement determines the required minimum HDS height at any time of measurement, either before or during the period of extended operation.

RAI-7

Regarding AI 7, there is new NRC staff guidance on the threshold limits for thermal embrittlement (TE) and irradiation embrittlement (IE) of Cast Austenitic Stainless Steel (CASS). The bases for the NRC staff’s new consensus on the threshold limits are described in “NRC position on Aging Management of CASS Reactor Vessel Internal Components” (ADAMS Accession Number ML14163A112).

a) Please address any difference between the new guidance and the evaluation performed for Salem, Units 1 and 2. In particular, please address, the new screening guidelines of CASS materials for loss of fracture toughness of highly irradiated components (i.e. components susceptible to IE), in addition to TE. If any changes to the evaluation are necessary, please submit the re-evaluation, if not, please explain why not. This evaluation could affect some of the CASS components that are listed as susceptible to TE in Tables 6-2 of Attachments 1 and 2 of the submittal for Salem, Units 1 and 2, respectively, especially the lower support column caps as described in the next paragraph.

b) The NRC staff's initial review indicated that, in addition to TE, the lower support column caps (which is one of the pieces that comprises the lower support column body as explained in Section 6.2.7 in Attachments 1 and 2 of the August 11, 2014, letter) are susceptible to IE. Please provide an explanation of how aging degradation due to TE and IE of the lower support column bodies is being managed and will be managed during the PEO.

Response to RAI-7

a) PSEG is currently working with the Electric Power Research Institute (EPRI) Industry Working Group and the PWROG to resolve the impact of the revised NRC staff thermal embrittlement threshold limits guidance (Reference 10) on the previous evaluations of RVI CASS component embrittlement. It is noted that previous evaluations, including those for Salem Units 1 and 2, have been performed using the screening process identified in MRP-191 (Reference 2), which formed the basis for the component rankings and recommended inspections identified in MRP-227-A (Reference 1). The Salem Units 1 and 2 responses to A/LAI 7 were based on TE threshold criteria identified in Reference 11, as recommended in Reference 1. The present process is less complex than that outlined in the revised NRC guidelines, but has slightly different and, in some cases, actually more conservative thresholds. The PWROG program is expected to deliver a resolution of this issue by the fourth quarter of the present calendar year. PSEG proposes to wait and then apply the approach to be identified in this resolution to readdress the assessments for potential thermal embrittlement of the CASS components employed in Salem Units 1 and 2. This reassessment will be accomplished in a timely manner after completion of the PWROG program, since the chemical compositions and ferrite contents of the CASS RVI components in Salem Unit 1 and Unit 2 have already been obtained and documented in the course of the previous evaluation.

b) The AMPs (References 4 and 5) for managing the aging degradation of the Salem Units 1 and 2 lower support column bodies were developed in compliance with the Inspection and Evaluation Guidelines, MRP-227-A (Reference 1). The aging degradation screening of the CASS lower support column bodies is recorded in MRP-191, Table 5-1 (Reference 2), which is a basis document for MRP-227-A. The CASS lower support column bodies were screened in as susceptible to IASCC, void swelling (VS) and both embrittlement mechanisms (TE and IE). The expert panel conclusions documented in MRP-191 (Reference 2) served as the basis for the aging management program guidelines provided in the Inspection and Evaluation Guidelines, MRP-227-A (Reference 1).

In MRP-227-A, Table 4-6 (Reference 1), the CASS lower support column bodies (of which the lower support column caps are a part) are listed as an Expansion component; they are linked to the Primary inspection component, the CRGT lower flange welds. The aging management plans developed for Salem Units 1 and 2 (References 4 and 5) followed the MRP-227-A (Reference 1) guidance for managing the potential embrittlement and cracking degradation of the lower support column caps at Salem Units 1 and 2 during the period of extended operation.

PSEG presently recognizes that MRP-191, Table 4-6 (Reference 2) identified that the CRGT lower flanges experience lower neutron fluence than the lower support column bodies. However, the industry is updating the guideline and is investigating an alternate primary component as the link for the lower core support column bodies as part of the revision to MRP-227-A. PSEG will implement the revised MRP-227 document when it is

issued and approved."

References

1. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.

2. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.

3. Westinghouse Technical Bulletin, TB-14-5, Rev. 0, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," August 25, 2014.

4. Westinghouse Report, WCAP-17397-NP, Rev. 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 1," July 2014. (ADAMS Accession Number ML14224A667).

5. Westinghouse Report, WCAP-17438-NP, Rev. 2, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem Unit 2," July 2014. (ADAMS Accession Number ML14224A667).

6. Salem Unit 1 Split Pin Replacement, NUCP 80088106, May 30, 2006.

7. Salem Unit 2 Split Pin Replacement, NUCP 80008845, November 11, 2001.

8. Westinghouse Report, WCAP-15028, Rev. 0, "Guide Tube Cold-Worked 316 Replacement Support Pin Development Program," March 1998 (Proprietary).

9. Westinghouse Report, WCAP-17096-NP, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009.

10. U.S. Nuclear Regulatory Commission Document, "NRC position on Aging Management of CASS Reactor Vessel Internal Components" (ADAMS Accession Number ML14163A112).

11. U.S. Nuclear Regulatory Commission Letter, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000 (NRC ADAMS Accession No. ML003717179).