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PSEG
Nuclear LLC

10 CFR 50.55a

LR-N11-0104
April 12, 2011

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

Salem Nuclear Generating Station, Unit 1
Facility Operating License No. DPR-70
NRC Docket No. 50-272

Subject: **Response to NRC Request for Additional Information regarding PSEG Relief Request S1-I4R-105**

References: (1) PSEG to NRC letter, "Request for Authorization to Continue using a Risk-Informed Inservice Inspection Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping," dated October 21, 2010.

In Reference 1, in accordance with 10 CFR 50.55a(a)(3), "Codes and standards," PSEG Nuclear LLC (PSEG) requested NRC approval of proposed Relief Request S1-I4R-105 for Salem Generating Station, Unit 1. The proposed relief will allow Salem to continue to utilize the NRC approved Salem Unit 1 Alternate Risk Informed Inservice Inspection (RI-ISI) program as an alternative to the 2004 Edition, ASME Section XI inspection requirements for specific Class 1 and Class 2 piping welds, in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety.

The NRC provided PSEG a Request for Additional Information (RAI) related to the Reference 1 request. Attachment 1 to this submittal provides the response to the RAI.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact me at 856-339-1466.

Sincerely,

Paul R. Duke, Jr.
Licensing Manager – PSEG Nuclear

Attachment

cc:

W. Dean - NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector – Hope Creek (X24)
P. Mulligan, Manager IV, NJBNE
Commitment Coordinator – Salem
PSEG Commitment Coordinator - Corporate

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO RELIEF REQUEST S1-I4R-105

FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

By letter dated October 21, 2010 (ADAMS Accession No. ML103060462), PSEG Nuclear LLC (PSEG, the licensee) submitted relief request S1-I4R-105 for Salem Nuclear Generating Station (Salem), Unit No. 1. The proposed relief would allow PSEG to continue using a risk-informed inservice inspection (RI-ISI) program as an alternative to the examination requirements specified in Section XI of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) for certain Class 1 and 2 piping welds.

The Nuclear Regulatory Commission (NRC) staff has reviewed the information the licensee provided that supports the proposed relief request and would like to discuss the following issues to clarify the submittal.

1. *Attachment 3 of the licensee's letter dated October 21, 2010, lists the following findings as gaps to the internal flooding requirements to the ASME Standard: IF-C1-01, IF-C2b-01, and IF-C2C-01. The resolution of these gaps state "See Appendix E of the Internal Flooding report. Very low risk areas were not addressed using the same level of detail as for higher risk areas." A quantitative guideline is not provided which would discern "low risk" from "high risk" areas as is provided in resolution of supporting requirement IF-C3b-01. Since a qualitative approach is assumed to be used for screening, please provide an explanation which shows that meeting IF-C1-01, IF-C2b-01, and IF-C2C-01 in entirety would not result in greater pipe segments for high or medium categories in the risk matrix.*

Response to Question 1:

The internal flood analysis did quantify all postulated scenarios that were not screened on the basis of Supporting Requirement (SR) IF-C5 of ASME/ANS RA-Sb-2005, which is now identified as IFSN-A12 per ASME/ANS RA-Sa–2009a, and also accounted for affected SSCs due to inter-area propagation pathways per IF-C3b (IFSN-A8 per ASME/ANS RA-Sa-2009a). For those quantified scenarios, a conditional core damage probability (CCDP) was computed that conservatively considered all PRA-modeled SSCs to be damaged by a flood originating or propagating into a particular flood area, which satisfied the ASME SR IF-C1 (IFSN-A1 per ASME/ANS RA-Sa-2009). When analyzing the various internal flood scenarios, credit was not initially given for area drainage and other mitigating effects in order to estimate a worst-case CCDP for that scenario. The CCDP was then multiplied by the flood initiating frequency to estimate the core damage frequency (CDF). If the CDF for a given flood scenario was sufficiently low, e.g., less than 0.1% of the nominal internal events CDF, then no further refinement was deemed necessary. However, if first estimates of the core damage frequencies for

that compartment proved too pessimistic, the affected area of the plant was analyzed in greater detail to take into account spatial effects, specific flooding flow rates, operator actions, drainage pathways, etc., which satisfied the considerations given in SRs IF-C2b and IF-C2c (IFSN-A4 and IFSN-A5 per ASME/ANS RA-Sa-2009, respectively). Therefore, the justification for more detailed modeling of certain flooding scenarios was aimed at removing some of the conservatism of the methodology, while at the same time providing a realistic basis for not assuming complete failure of all scenario-specific equipment due to a credible flooding event.

In reviewing the CDF results for the flood scenarios reported in the Internal Flood Analysis documentation, many of the scenarios were less than a tenth of a percent of the nominal CDF value. Collectively, these scenarios contributed a sum total of less than 0.6% of the nominal CDF and were excluded from any further detailed analysis, which would have invoked the mitigative measures allowed per SRs IF-C2b and IF-C2c. Therefore, those internal flood scenarios where detailed modeling was not performed were not "screened" from quantification. These scenarios were merely excluded from any further detailed analysis that would involve hydraulic modeling of flood areas, which would involve development of specific flooding flow rates and height of water as a function of time for various flood areas. There was little benefit to be gained from performing time-intensive detailed evaluations of scenarios that proved to be relatively insignificant in comparison to other internal flood scenarios.

2. *Attachment 2 of the licensee's letter dated October 21, 2010, shows the number of welds to be examined in the fourth interval will decrease. However, the NRC staff cannot determine if the locations to be examined have changed. Are the inspection locations in the RI-ISI program that have been developed for the fourth 10-year interval the same locations as those in the third interval RI-ISI program approved in the NRC staff's October 1, 2003, safety evaluation? If not, please summarize the changes to the program and what caused those changes.*

Response to Question 2:

The majority of the RI-ISI inspection locations for the fourth 10-year interval are identical to those in the third 10-year interval. The RI-ISI methodology does not require that the same selections be maintained for subsequent intervals, or even within the same interval, but only requires that sufficient selections be maintained to meet all requirements of the methodology. The RI-ISI being a living program, changes in the inspection locations may result due to:

- Plant modifications impacting the total number of RI-ISI piping welds
- RI-ISI piping welds required to be included in other Augmented inspection programs (i.e. MRP-139)
- PRA updates impacting the consequence category
- Degradation mechanism assignment changes (due to improved or updated information) impacting the failure potential category
- Improved or updated information regarding radiation exposure, access, inspectability or other factors

3. *Has the percentage of Class 1 welds to be examined remained at approximately 8.5% for the fourth 10-year interval RI-ISI program?*

Response to Question 3:

Yes, the percentage of welds to be examined for the fourth interval has remained effectively constant at approximately 8.5%.