April 29, 2005

Mr. William Levis
Senior Vice President & Chief Nuclear Officer
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 2, EVALUATION OF RELIEF REQUEST S2-I3-RR-A06 (TAC NO. MC6667)

Dear Mr. Levis:

By letter dated April 8, 2005, PSEG Nuclear, LLC (PSEG) submitted Relief Request S2-I3-RR-A06 for the Salem Nuclear Generating Station, Unit No. 2 (Salem 2). The request was submitted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i) as a proposed alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The submittal proposed implementation of the alternative requirements of ASME Code Case-566-2 for evaluation of bolted connections when leakage occurs.

Accordingly, the Nuclear Regulatory Commission (NRC) staff has reviewed the request against the requirements of 10 CFR 50.55a as related to the implementation of ASME Code, Section XI. Based on the information provided, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative for use of ASME Code Case-566-2, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) until such time as the ASME Code Case is published in a revision to Regulatory Guide 1.147. The NRC staff's review of this Relief Request is documented in the enclosed Safety Evaluation (SE).

The NRC staff notes that your submittal of this Relief Request was not timely in that it was submitted after the Salem 2 2005 refueling outage had begun and the request was not the result of conditions or issues identified after shutdown. Indeed, the staff notes that a draft version of this relief request was available in October 2004. Late submission of this request required reallocation of NRC staff resources from other ongoing work in order to support the Salem 2 outage schedule. Furthermore, because the submittal was not timely, this relief request would not have been a suitable candidate for the NRC staff to provide verbal authorization for its implementation, had PSEG requested such consideration.

If you have any questions regarding this SE, please contact the Salem Project Manager, Daniel Collins at (301) 415-1427.

Sincerely,

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-311

Enclosure: As stated

cc w/encl: See next page

If you have any questions regarding this SE, please contact the Salem Project Manager, Daniel Collins at (301) 415-1427.

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/RA/

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUEST TO IMPLEMENT REQUIREMENTS OF ASME CODE CASE N-566-2

PSEG NUCLEAR, LLC

SALEM GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated April 8, 2005, PSEG Nuclear, LLC, (PSEG or the licensee) requested relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements at Salem Nuclear Generating Station, Unit No. 2 (Salem 2). Relief Request (RR) S2-I3-RR-A06 proposes to use ASME Code Case N-566-2 which is an alternative to the requirements of ASME Code, Section XI, sub-paragraph IWA-5250 (a)(2) for Class 1, 2, and 3 bolted connections.

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of the ASME Code Class 1, Class 2, and Class 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g), except where specific written relief has been granted by the Nuclear Regulatory Commission (NRC or the Commission) pursuant to 10 CFR 50.55a(g)(6)(i) or alternatives are authorized pursuant to 10 CFR 50.55a(a)(3). Section 50.55a(a)(3) of 10 CFR states, in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examinations of components and system pressure tests conducted during the first 10-year ISI interval, and subsequent intervals, comply with the

Enclosure

requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval. The ISI code of record for the third 10-year ISI interval for Salem 2 is the 1998 Edition through the 2000 Addenda.

3.0 TECHNICAL EVALUATION - RR SC-I3-RR-A06

3.1 System/Components for Which Relief is Requested

ASME Code Section XI, 1998 Edition through the 2000 Addenda, Class 1, 2, and 3 bolted connections.

3.2 ASME Code Requirements

ASME Code Section XI,1998 Edition, 2000 Addenda, sub-paragraph IWA-5250(a)(2), "Corrective Action," requires that the sources of leakage detected during the conduct of a system pressure test shall be located and evaluated by the Owner for corrective action as follows:

> If leakage occurs at a bolted connection in a system borated for the purpose of controlling reactivity, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100. The bolt selected shall be the one closest to the source of leakage. When the removed bolt has evidence of degradation, all remaining bolting in the connection shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

3.3 Proposed Alternative and Basis for Use

The licensee requested relief from the ASME Code required corrective action described in sub-paragraph IWA-5250(a)(2) and, as an alternative, proposes to use ASME Code Case N-566-2. The licensee requested that relief be granted for the inservice examinations to be performed during the third 10-year ISI interval. The RR pertains to ASME Code Section XI, Class 1, 2, and 3 bolted connections. In the licensee's proposed alternative to use ASME Code Case N-566-2, the licensee also made the following statements:

> In the event that the evaluation of ASME Code Case N–566-2 paragraphs (a) or (b) determines that the relevant condition can be accepted for continuing service, PSEG would periodically conduct visual inspections of the affected components (in accordance with PSEG's ASME [Code] Section XI Repair Program requirements) to verify operating conditions have not changed affecting earlier calculation assumptions.

> In the event that the relevant condition cannot be accepted for continuing operation using ASME Code Case N–566-2 paragraphs (a) or (b), Salem Generating Station Unit 2 would correct the condition in accordance with its ASME [Code] XI Article IWA-4000 Repair/Replacement Activities and the PSEG ASME [Code] Section XI Repair Program Requirements.

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief on the basis that the proposed alternative provides an acceptable level of quality and safety. The licensee's basis, in part, for its alternative providing an acceptable level of quality and safety is as follows:

> Removal of bolts for VT-3 visual examination is not always the most prudent action when leakage is discovered at a bolted connection. Leakage at bolted connections is typically identified during system leakage tests. For Class 1 systems, this leakage test is conducted prior to plant startup following each refueling outage. This test is performed at full operating pressure (2235 psig) and temperature. When leakage is discovered during this test, the corrective action (i.e., removal of bolts) must be performed with the system at full temperature and pressure, or the plant must be cooled down. The removal of a bolt at full temperature and pressure conditions can be extremely physically demanding due to the adverse heat environment. Cooling down the plant subjects the plant to additional heatup and cool down cycles, and can add 3-4 days to the duration of an outage. Bolted connections associated with pumps and valves are typically studs threaded into the body of the component. Removal of these studs is typically very difficult, requiring expenditure of both time and dose resources due to length of time they have been installed and are often damaged during the removal process. This difficulty is compounded when the removal must be performed under heat stress conditions.
>
> The requirements of IWA-5250(a)(2) must be applied regardless of the significance of the leakage or the corrosion resistance of the materials used in the bolted connection. Implementation of [ASME] Code Case N-566-2 requires factors such as the number and service age of the bolts, the bolting materials, the corrosiveness of the system fluid, the leakage location and system function, leakage history at the connection or at other system components, and visual evidence of corrosion at the bolted connection be used to evaluate the need for corrective measures.

3.4 Staff Evaluation

IWA-5250(a)(2) requires that one bolt be removed from a leaking bolted connection and that the bolt be VT-3 visually examined for corrosion and evaluated in accordance with IWA-3100. The bolt selected must be the one closest to the source of leakage. If the bolt shows evidence of degradation, all remaining bolting in the connection shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100. The ASME Code requirements provide assurance that bolting corroded by system leakage will be detected and that corrective actions will be taken. However, the ASME Code requirements may be overly conservative since the removal and examination of all bolting may not be necessary to assure continued integrity of a bolted connection. Furthermore, corrosion in the joint region may depend on other factors beyond leakage. Thus, in the instances where leakage has been identified at bolted connections, the requirements of Section XI of the ASME Code do not always provide for the most reasonable course of action.

As an alternative to the requirements of IWA-5250(a)(2), ASME Code Case N-566-2 requires that either (a) or (b), as stated below, is met:

(a) The leakage shall be stopped, and the bolting and component material shall be evaluated for joint integrity as described in paragraph (c) below.

(b) If the leakage is not stopped, the owner shall evaluate the structural integrity and consequences of continuing operation, and the effect on the system operability of continued leakage. This engineering evaluation shall include the considerations in (c) below.

(c) The evaluation of (a) and (b), above, is to determine the susceptibility of the bolting to corrosion and failure. This evaluation shall include the following:

1. Number and service age of the bolts
2. Bolt and component material
3. Corrosiveness of process fluid
4. Leakage location and system function
5. Leakage history at the connection or other system components
6. Visual evidence of corrosion at the assembled connection

The staff considers this to be a reasonable approach, consistent with the rest of the ASME Code, in evaluating components for continued service. The licensee's statement that it will periodically conduct visual inspections of the affected components (in accordance with PSEG's ASME Code Section XI Repair Program requirements) to verify operating conditions have not changed that would affect earlier calculation assumptions, will provide further assurance of structural integrity. Based on the discussion above, the staff concludes that the use of ASME Code Case N-566-2 under the alternative proposed in RR SC-I3-RR-A06 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the proposed alternative for the third 10-year ISI interval at Salem 2.

3.5 Conclusion

The NRC staff has evaluated the licensee's request to implement alternatives to certain ASME Code requirements, as contained in ASME Code Case N-566-2. The staff has determined that implementation of the requirements of this ASME Code Case will continue to provide an acceptable level of quality and safety. Therefore, the alternatives proposed in RR SC-I3-RR-A06 are authorized for use, pursuant to 10 CFR 50.55a(a)(3)(i). Use of ASME Code Case N-566-2 is authorized until such time as the ASME Code Case is published in a revision to Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement this AMSE Code Case, the licensee must follow all provisions in Code Case N-566-2 with limitations issued in Regulatory Guide 1.147, if any. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this RR remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: R. Davis

Date: April 29, 2005