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MAY 0 5 2009

LR-N09-0103

PSEG
Nuclear LLC

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

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

Subject: 2008 Summary of Revised Regulatory Commitments

In accordance with the Nuclear Energy Institute (NEI) process for managing Nuclear Regulatory Commission (NRC) commitments and associated NRC notifications, PSEG Nuclear LLC (PSEG) submits this correspondence to discuss commitments that were changed and not reported by other means during 2008.

The attached commitments were evaluated in accordance with the requirements of PSEG's Regulatory Commitment Change Process, which is consistent with the guidance in NEI 99-04, "Guideline for Managing NRC Commitments." Additional documentation is available for your review.

There are no new commitments in this letter.

If there are any questions, please contact Howard Berrick at 856-339-1862.

Sincerely,

Robert C. Braun
Site Vice President - Salem

Attachments (1)

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NRR

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	Revised Commitment Description	Justification For Change
Original Commitment: The design and fabrication of the Outage Equipment Hatch (OEH) at Salem 1 and 2 is in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, 1989 edition or later. **Source Document:** Salem Unit 1/2 Amendments 217/199 and 263/245 Safety Evaluation Reports (SER) (as corrected by letters LR-N04-0359 dated March 11, 1999, and LR-N07-0288 dated November 19, 2007). **Reference:** CR 80094905 **Date of Change:** 03/01/08	Equivalent closure devices for the equipment hatch in accordance with Technical Specification (TS) 3/4.9.4 will be designed and fabricated in accordance with commercial grade requirements to serve as a ventilation barrier and are not required to be designed and fabricated in accordance with ASME Section VIII. As stated in Salem TS 3/4.9.4 bases, 'any equivalent closure device used to satisfy the requirements of TS 3/4.9.4.a will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident (FHA) during refueling operations are met." The closure of the ventilation barrier is to provide additional defense-in-depth by closing the equipment hatch release pathway although the FHA analysis performed using the AST, approved by Amendment 251/232, assumed a 100% release over 2 hours through the open equipment hatch. Since pressurization events are unlikely during MODE 6 (refueling) operation, the closure requirement for the equivalent device need only be sufficient to provide an atmospheric ventilation barrier to restrict radioactive material released from a fuel element rupture during refueling operation.	As approved in Amendments 251/232 the Fuel Handling Accident (FHA) analysis was revised using the Alternate Source Term (AST) methodology. The revised AST FHA assumed a 100% release over a 2-hour period through the open equipment hatch. During the approval of Amendments 263/245 to allow the containment equipment hatch to be open during the movement of irradiated fuel within the containment, PSEG committed to establishing the closure of the containment equipment hatch within 1-hour following an FHA to provide additional defense in depth. In 1998 when the Nuclear Regulatory Commission (NRC) approved Amendments 217/199 to allow an equivalent closure device to be used in lieu of the inner equipment hatch, PSEG stated that this equivalent closure device would be designed in accordance with the ASME B&PV Code, Section VIII, 1989 edition or later. As stated in the SER for Amendments 217/199 and the current Salem TS 3/4.9.4 basis, 'any equivalent closure device used to satisfy the requirements of TS 3/4.9.4.a will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met." Prior to the approval of the AST analysis for the FHA on October 10, 2002, the closure of the containment was performed to mitigate the consequences of the FHA by limiting the release pathway. However, with the implementation of the FHA AST analysis, closure of the containment is no longer an action to mitigate the consequences of the FHA. Although it is not credited in the mitigation of the FHA, PSEG committed in the approval of Amendments 263/245 to have the ability to close the containment within 1-hour following an FHA to add additional defense in depth. As was recognized in the SER for Amendments 217/199, pressurization events are unlikely during Mode 6 (refueling) operations; the closure requirement for the equivalent device need only be sufficient to provide an atmospheric ventilation barrier.

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		Calculation S-2-RC-MDC-2151 was performed to evaluate the containment environment during modes 5 and 6 following a loss of RHR event. This calculation concluded that only an atmospheric-pressure closure device (ventilation barrier) is necessary to meet the containment closure commitment following a FHA. The closure device can be fabricated from metal or flexible materials that will prevent gross ventilation flow through the equipment hatch. It is capable of being closed within one hour (or prior to core boiling) and is designed to accommodate the tracks and other materials that may be installed through the hatch. In view of the above, an equivalent closure device need only be an atmospheric ventilation barrier to restrict radioactive material release. In the case of the FHA, containment pressure would be atmospheric pressure to the extent that an equivalent barrier would not need to resist internal containment pressure.

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Original Commitment: PSEG will use the NRC approved methodology in WCAP-11394-P-A for each fuel cycle to ensure the minimum departure from nucleate boiling ratio (DNBR) is maintained above the DNBR safety limit. **Source Document:** LR-N06-0035 NRC Amendment SER 278/261 dated March 19, 2007 Salem Updated Final Safety Analysis Report (UFSAR), Chapter 15 **Reference:** 70085988 **Date of Change:** 09/11/08	PSEG is deleting the explicit commitment to WCAP-11394.	WCAP-11394 is not a Salem specific report, but a generic Westinghouse topical report that is an inherent part of the Westinghouse reload safety analysis process. Per Westinghouse, this WCAP is referenced in their dropped rod protection analysis guidance, which is utilized for Salem. The WCAP is also referenced in the Salem UFSAR Section 15.2.3, "the transient response analysis, nuclear peaking factor analysis, and performance of the departure from nucleate boiling (DNB) design basis confirmation are performed in accordance with the methodology described in Reference 15 [WCAP-11394]. Note that the analysis does not take credit for the power-range negative flux rate trip." Salem implemented this WCAP as the dropped rod event design basis in the 1990s and from that time has not taken credit for the negative flux rate trip (NFRT) in the accident analyses. The current PSEG Nuclear procedures associated with core reload and vendor reload analyses are: NF-AA-100 (Reload Control Procedure), NF-AA-100-1001 (Core Reload and Cycle Management Configuration Changes Using SAP), and NF-AP-100-7000 (Westinghouse NSSS Reload Design Control Implementation). In addition PSEG maintains a formally signed interface agreement with Westinghouse (NFS-0180, "Reload Design Interface Document for Salem Units 1 and 2"). These documents do not include any specific list of Chapter 15 events that are evaluated on a reload basis, or any specific references to Westinghouse codes, methods, or topical reports. However, the procedures and interface documents do cover specific reload milestones and information transmittals affecting the accident analyses. These include the Reload Safety and Licensing Checklist (RS&LC) and Reload Safety Analysis Checklist (RSAC). The RS&LC is a document the customer prepares and sends to Westinghouse that covers any anticipated plant, operating and licensing changes to be implemented for the upcoming cycle. The RS&LC is included along with other reload documentation in a

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		transmittal referred to as the Reload Design Initialization (RDI). Westinghouse provides the RSAC to the customer; it lists those parameters that most significantly affect the accident analysis for the reload design, along with the current limits utilized in the analyses. Implementation of any new Westinghouse method is specifically called out in the above procedures: NF-AA-100 Step 4.83, "if required then ensure a 50.59 review is performed for any identified changes such as methodology changes..."; NF-AP-100-700 Step 4.8.10, "review and comment to Westinghouse on the draft Safety Assessment and RSAC current limits. Request calculations from Westinghouse, and review for (1) any non-routine analyses, (2) any analyses being performed for the first time, and (3) and analyses that had a methodology change."; NF-AP-100-700 Attachment 3 (Reload Design Finalization Checklist), #8, "are all methods and models used in the reload design fully accepted by PSEG and the NRC"; NFS-0180, "methods and computer code for nuclear design are those developed or adapted by Westinghouse. These are described in the methods, codes and design manuals. Any exceptions to this use must be agreed upon and documented." Although the commitment to use WCAP-11394 (for the dropped rod event supporting NFRT removal) is not explicitly included in the Nuclear Fuels reload procedures, it is an inherent part of the Westinghouse accident analysis design basis. Since the above procedures and interface document with Westinghouse explicitly cover any changes to their methods that can affect Salem reload design and licensing requirements, the necessary commitment is already implicitly covered and there is no need for any additional explicit commitment in the noted Nuclear Fuel reloads design procedures.

	Revised Commitment Description	Justification For Change
Original Commitment: PSEG to provide "...a complete formal training program be implemented for all the mechanical and electrical maintenance, quality control, and operating personnel, including supervisors who will be responsible for the maintenance and availability of the diesel generators. The depth and quality of this training program shall be at least equivalent to that of training programs normally conducted by major diesel engine manufacturers." This commitment was implemented. **Source Document:** Salem Unit 2 SER 8.3.4 (d) Supplement 5 **Reference:** 70077701 **Date of Change:** 03/01/08	PSEG provided an "initial" complete formal training and a qualification program for Emergency Diesel Generator (EDG) maintenance for applicable personnel who work on and supervise EDG maintenance activities. Continuing (future) training will be decided using the systems approach to training IAW 10CFR 50.120 and the training process description.	The present training program is covered under the systems approach to training [as defined in 10 CFR 55.4] as outlined in 10 CFR 50.120 (b). The training programs incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility.