February 27, 2008

Mr. William Levis
President & Chief Nuclear Officer
PSEG Nuclear LLC - N09
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: STEAM GENERATOR FEEDWATER PUMP TRIP, FEEDWATER ISOLATION VALVE RESPONSE TIME TESTING AND CONTAINMENT COOLING SYSTEM (TAC NOS. MD4843 AND MD4844)

Dear Mr. Levis:

The Commission has issued the enclosed Amendment Nos. 287 and 270 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. These amendments consist of changes to the Updated Final Safety Analysis Report (UFSAR), the Technical Specifications (TSs) and the facility operating licenses in response to your application dated March 16, 2007, as supplemented by letters dated August 30, September 14, and November 20, 2007, and January 16, 2008.

The amendments revise the UFSAR to modify the Salem licensing basis with respect to the response times associated with a steam generator feedwater pump (SGFP) trip and feedwater isolation valve (FIV) closure. The amendments also revise the TS requirements for the containment fan cooler unit (CFCU) cooling water flow rate. These changes are associated with a revised containment response analysis that credits an SGFP trip and FIV closure (on a feedwater regulator valve failure) to reduce the mass/energy release to the containment during a main steam line break. The containment analysis also credits a reduced heat removal capability for the CFCUs, allowing a reduction in the required service water flow to the CFCUs.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/ra/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 287 to License No. DPR-70
2. Amendment No. 270 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/ra/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
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Amendment Accession No: ML080220404
TS Accession Nos.: ML080220414 (Unit 1); ML080220424 (Unit 2)

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PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 287
License No. DPR-70

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees) dated March 16, 2007, as supplemented by letters dated August 30, September 14, and November 20, 2007, and January 16, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 287 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to restart from refueling outage 1R19. Implementation shall include revision of Updated Final Safety Analysis Report Table 7.3-8 as shown in the licensee's letter dated January 16, 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and the Technical Specifications

Date of Issuance: February 27, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 287

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of Facility Operating License No. DPR-70 with the attached revised page as indicated. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 4	Page 4

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3/4 6-11a	3/4 6-11a

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 270
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees) dated March 16, 2007, as supplemented by letters dated August 30, September 14, and November 20, 2007, and January 16, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 270, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to restart from refueling outage 2R16. Implementation shall include revision of Updated Final Safety Analysis Report Table 7.3-9 as shown in the licensee's letter dated January 16, 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and the Technical Specifications

Date of Issuance: February 27, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 270

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of Facility Operating License No. DPR-70 with the attached revised page as indicated. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 4	Page 4

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3/4 6-13	3/4 6-13

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 287 AND 270 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By application dated March 16, 2007, as supplemented by letters dated August 30, September 14, and November 20, 2007, and January 16, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML070871102, ML072540626, ML072680833, ML073330456, and ML080460105 respectively), PSEG Nuclear LLC (PSEG or the licensee) submitted an amendment request for Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The proposed amendment would revise the Updated Final Safety Analysis Report (UFSAR) to modify the Salem licensing basis with respect to the response times associated with a steam generator feedwater pump (SGFP) trip and feedwater isolation valve (FIV) closure. The amendment would also revise the Technical Specification (TS) requirements for the containment fan cooler unit (CFCU) cooling water flow rate. These changes are associated with a revised containment response analysis that credits an SGFP trip and FIV closure (on a feedwater regulator valve (FRV) failure) to reduce the mass/energy release to the containment during a main steam line break (MSLB). The containment analysis also credits a reduced heat removal capability for the CFCUs, allowing a reduction in the required service water (SW) flow to the CFCUs.

The supplements dated August 30, September 14, and November 20, 2007, and January 16, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 10, 2007 (72 FR 17951).

2.0 REGULATORY EVALUATION

The licensee addressed the regulatory requirements applicable to the proposed amendment in Section 5.2 of Attachment 1 to the application dated March 16, 2007. The regulatory

Enclosure

requirements, criteria, and guidance which the NRC staff applied in its review are discussed below.

As discussed in Section 3.1 of the Salem UFSAR and Section 5.2 of Attachment 1 to PSEG's application dated March 16, 2007, the general design criteria (GDC) followed in the design of Salem Units 1 and 2 are the Atomic Industrial Forum version, as published in a letter to the Atomic Energy Commission from E. A. Wiggin, Atomic Industrial Forum, dated October 2, 1967. PSEG's application dated March 16, 2007, states that the Salem Unit 1 and 2 design also conforms to the intent of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), "General Design Criteria for Nuclear Power Plants," dated July 7, 1971. The following GDCs in Appendix A to 10 CFR Part 50 are applicable to the proposed amendment:

- GDC 16, "Containment design," insofar as it requires that the containment and its associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- GDC 19, "Control room," insofar as it requires that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions.

- GDC 38, "Containment heat removal," insofar as it requires that the reactor containment be provided with a system to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA), and maintain them at acceptably low levels.

- GDC 50, "Containment design basis," insofar as it requires that the containment structure, including access openings, penetrations and the containment heat removal system be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," specifies requirements for periodic testing of the leak tightness of the containment and its penetrations. Appendix J includes requirements for three types of tests. Type A tests are intended to measure the overall integrated leakage rate of the containment. Type B tests are intended to measure leakage for certain types of containment penetrations (e.g., penetrations whose design includes resilient seals, gaskets, or sealant compounds). Type C tests are intended to measure containment isolation valve leakage rates. Appendix J includes two options (Option A or Option B) for meeting the requirements of the appendix. Option A provides prescriptive requirements while Option B provides a performance-based approach. As discussed in TS 6.8.4.f, the Salem Primary Containment Leakage Rate Testing Program utilizes Option B to Appendix J for the Type A, B, and C testing.

Requirements for the design and analysis of emergency core cooling systems (ECCS) are specified in 10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." These regulations are in place to ensure adequate core cooling following a LOCA such that

certain acceptance criteria are satisfied. As specified in 10 CFR 50.46(b), the acceptance criteria include peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

Licensees are required by 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," to establish a program for qualifying electrical equipment important to safety for the environmental conditions that may be present at the location where the equipment must perform its function. This provides assurance that the equipment needed during and following a design-basis event will perform its intended functions.

Previously, in Amendments 271 and 252, issued February 17, 2006 (ADAMS Accession No. ML060040322), Salem Units 1 and 2 implemented of a full-scope alternative source term (AST) in accordance with 10 CFR 50.67, "Accident source term," and following the guidance of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The NRC staff's evaluation of this proposed amendment has been conducted to verify that the results of the licensee's affected design-basis accident (DBA) radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67 for offsite doses and GDC 19 "Control room." The applicable acceptance criteria are 5 rem total effective dose equivalent (TEDE) in the control room, 25 rem TEDE at the exclusion area boundary, and 25 rem TEDE at the outer boundary of the low population zone. Except where the licensee proposed a suitable alternative, the staff used the guidance provided in applicable sections of RG 1.183 and NUREG-0800, Standard Review Plan (SRP), Chapter 15, "Accident Analysis" and SRP Section 6.4, "Control Room Habitability System," in performing this review.

NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions" dated September 30, 1996, requested that licensees determine: (1) if containment air cooler cooling water systems are susceptible to either water-hammer or two-phase flow conditions during postulated accident conditions; and (2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that over-pressurization of piping could occur. The NRC staff evaluated the proposed amendment with respect to any potential impact on the prior resolution of the issues discussed in GL 96-06 for Salem Units 1 and 2.

On July 18, 1989, the NRC staff issued GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," due to continuing problems of bio-fouling of SW systems. For plants with open-cycle SW systems (such as Salem Units 1 and 2), the GL recommended implementation of an ongoing program of surveillance and control techniques to significantly reduce the incidence of flow blockage problems as a result of bio-fouling. The NRC staff evaluated the proposed amendment with respect to any potential impact on the prior resolution of the issues discussed in GL 89-13 for Salem Units 1 and 2.

In addition, the NRC staff's evaluation used the following guidance: SRP Section 6.2.1, "Containment Functional Design" and SRP Section 6.2.2, "Containment Heat Removal Systems."

3.0 TECHNICAL EVALUATION

As discussed in Section 2 of Enclosure 1 of the licensee's application dated March 16, 2007, as part of an effort referred to as the "CFCU Margin Recovery Project," PSEG proposed to demonstrate reduced reliance on CFCU cooling during accident conditions. Lower CFCU heat removal rates would potentially allow future planned modifications or analysis revisions such as increased plugging of CFCU heat exchanger tubes, justification of increased tube fouling, and/or allow a reduction of SW accident flow rates. As part of this effort, Westinghouse performed a revised containment response analysis as documented in WCAP-16503-NP, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit Margin Recovery Project," dated February 2007 (Reference 7). The analysis credits a reduced heat removal capability for the CFCUs, allowing a reduction in the required SW flow rate to the CFCUs. The analysis also credits a SGFP trip and FIV closure to reduce the mass and energy release to containment for an MSLB.

As shown in Salem UFSAR Figures 10.4-5A and 10.4-5B, each Salem unit contains two SGFPs and four steam generators (SGs). The SGFPs supply main feedwater (MFW) through four lines penetrating the containment wall, one line feeding each SG. Each of the four MFW lines contains an FIV (the BF-13 valves), an FRV (the BF-19 valves), and an FRV bypass valve (the BF-40 valves), all located outside containment. Following an MSLB, MFW flow is isolated to limit the mass and energy released into the containment through the failed steam line. Isolation of MFW is accomplished via closure of the FRVs and FRV bypass valves, trip of the SGFPs, and closure of the FIVs.

As described in UFSAR Section 6.2, post-accident heat removal capability for the containment is provided by two separate, engineered safety features systems. They are the containment spray system and the containment fan cooling system. The containment fan cooling system consists of five CFCUs. Each CFCU is designed for two speeds, operating at high speed during normal operation and low speed during accident conditions. Each CFCU is provided with approximately 1000 gallons per minute (gpm) of SW during normal operation with flow increasing to approximately 2500 gpm during conditions of safety injection (SI).

The current SW flow path design to the CFCUs consists of a dual-mode flow control with three active air-operated control valves in each of the five CFCU flow paths. The licensee stated that each of these valves must change state on an accident signal and the CFCU outlet control valve must switch between two different flow control set points (normal/low flow to accident/high flow). The licensee desires to simplify the current design. The licensee's application indicated that revised containment analysis, performed to support the proposed amendment, shows that credit for the heat removal capability for the CFCUs can be reduced. This would allow an SW flow lower than the current accident design but still meeting the normal operating requirements, thus eliminating the need for complex controls and allowing simplification of the design.

The current Salem Unit 2 SGs (Westinghouse Model 51) are scheduled to be replaced by Framatome ANP Model 61/19T SGs during refueling outage 2R16 in spring, 2008. The results of the revised containment analysis indicated that Unit 2, with the replacement steam generators (RSGs), would be the most limiting plant and configuration.

The current Salem licensing basis for an MSLB is the Westinghouse steam line break mass and energy release methodology as documented in WCAP-8822, “Mass and Energy Releases Following a Steam Line Rupture” (Reference 5). The Westinghouse methodology is approved by the NRC (Reference 6). The governing case in determining the maximum containment pressure and temperature is an MSLB, with a single failure of the FRV on the faulted loop SG. As stated in UFSAR Section 15.4.8.2.2, the FRVs close within 10 seconds of any SI signal and the FIVs close within 32 seconds of any SI signal. Thus, in the current analysis, failure of the FRV to close results in an additional 22 seconds during which pumped MFW from the Condensate/Feed system may be added to the faulted SG. Also, the current containment analyses did not credit the reduced mass and energy release that would occur with the SGFP trip, even though it is allowed by the approved MSLB accident methodology.

The combined need of the CFCU margin recovery and the Unit 2 RSGs has resulted in the licensee’s proposal for adoption of the new containment response analysis. The revised analysis will support Salem Units 1 and 2 with the current SGs and Unit 2 with the RSGs. The revised analyses take credit for the SGFP trip and incorporate other less restrictive input assumptions to reduce the mass and energy release to containment during an MSLB.

As described in PSEG’s application dated March 16, 2007, the proposed amendment would revise TS Table 3.3-5, TS 4.6.2.3.b.3, TS 4.6.2.3.c.2 for both Salem Units 1 and 2. However, on June 19, 2007, the NRC issued Amendment 283 (for Salem Unit 1) and Amendment 266 (for Salem Unit 2) which, in part, relocated TS Table 3.3-5 for Salem Unit 1 to UFSAR Table 7.3-8 and TS Table 3.3-5 for Salem Unit 2 to UFSAR Table 7.3-9. PSEG’s supplement dated January 16, 2008 (Reference 13), provided a mark-up of UFSAR Table 7.3-8 and 7.3-9 consistent with the previous changes proposed to TS Table 3.3-5 in the application dated March 16, 2007 (as modified by the licensee’s letter dated September 14, 2007). PSEG stated that the changes to UFSAR Tables 7.3-8 and 7.3-9 would require prior NRC approval in accordance with 10 CFR 50.59(c)(2)(viii) (i.e., proposed changes would result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses). As such, the proposed amendment would include the following changes:

1) TS Surveillance Requirements (SRs) 4.6.2.3.b.3 and 4.6.2.3.c.2 establish the requirements for verifying the cooling water flow rate to the CFCUs. The flow rates would be changed from the current value of 2550 gpm to 1300 gpm to reflect the revised containment analysis.

2) UFSAR Table 7.3-8, “Salem Unit 1 Engineered Safety Features Response Times” and UFSAR Table 7.3-9, “Salem Unit 2 Engineered Safety Features Response Times” would be revised to add new requirements for the response times associated with a SGFP trip and FIV closure consistent with the revised containment analysis. Currently, the response time for feedwater isolation shown in the tables, for all initiating signals, only includes the 10 second response time for the FRV and FRV bypass valve closure. The response times (for the initiating signals that are required by the revised containment response analysis) for the SGFP trip and the FIV closure will be added to UFSAR Table 7.3-8 and 7.3-9 by replacing the current values for feedwater isolation response time with reference to a new note (Note 8). The new note would read as follows:

"Feedwater isolation includes closure of the feedwater regulating valves (FRV), the FRV bypass valves, the feedwater isolation valves (FIV), and trip of the steam generator feedwater pumps (SGFP). The response time for feedwater isolation by closure of the FRVs (the BF-19 valves) and the FRV bypass valves (the BF-40 valves) is 10 seconds. The response time for feedwater isolation by closure of the FIVs (the BF-13 valves) is 32 seconds. The response time for feedwater isolation by trip of the SGFPs is 7 seconds, not including pump coastdown time."

An additional note (Note 9) would be added to UFSAR Tables 7.3-8 and 7.3-9 for the initiating signals for feedwater isolation that are not impacted by the proposed change, clarifying that their response time only credits closure of the FRVs (BF-19 valves) and FRV bypass valves (BF-40 valves).

The NRC staff's technical evaluation of the proposed amendment is organized as follows:

3.1 Containment Analysis
3.2 Containment Analysis Results
3.3 Conclusions Regarding Containment Analysis
3.4 Radiological Dose Consequences
3.5 Environmental Qualification
3.6 CFCU Performance
3.7 Overpressurization of Piping Penetrating Containment
3.8 Core Cooling
3.9 Technical Evaluation Conclusion

3.1 Containment Analysis

The LOCA and MSLB cases for Salem Units 1 and 2 were reanalyzed in WCAP-16503-NP (Reference 7) to address the long-term LOCA and worst case MSLB scenarios. The methods used in the analysis incorporate modeling refinements consistent with the current Salem licensed methodology for LOCA and MSLB containment response. The current and the revised analyses have used the same codes for mass and energy releases, LOFTRAN (Reference 8) for MSLB, and SATAN-VI for LOCA. The long-term mass and energy releases for LOCA were calculated by the codes WREFLOOD for core reflooding phase, FROTH for post-reflood, and EPITOME to cover the period from the time the secondary pressure equilibrates to the saturation temperature at the containment design pressure to the steam generator depressurization. Both the current and revised analyses have used the code COCO for containment pressure and temperature calculations (Reference 9).

The analyses considered a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the MSLB mass and energy releases for use in containment analyses. In order to capture the influence of all factors that effect the quantity and rate of the mass and energy release, a spectrum of cases were analyzed that vary the initial power level, the break type, the break area, the break location, and the single failure. The postulated single failures are the containment safeguards failure (CSF), auxiliary feedwater runout protection failure, FRV failure, and main steam isolation valve (MSIV) failure. The licensee's proposed change placed a significant emphasis on the FRV failure. The key assumptions of WCAP-16503-NP for an MSLB with a single failure of the FRV (failing

open), which differ from the current analysis, are summarized below (all times indicated are after reaching SI set point):

- The current analysis did not take credit for SGFP trip. In the revised analysis, the SGFPs trip in 7 seconds and coastdown in an additional 7 seconds. However, the condensate pumps do not trip and can continue to provide pumped flow when the faulted SG depressurizes until the feedwater flow path is isolated.

- The current analysis assumes the FIV on the faulted loop closes in 32 seconds, with a decrease in the MFW flowrate over the last 20 seconds of the 30-second stroke time of the FIV. In WCAP-16503-NP, the FIV also closes in 32 seconds, but with a reduction in MFW flowrate over the last 10 seconds of the 30-second stroke time.

- The CFCU heat removal rates were lowered in the revised analysis, to reflect the desired reduction in SW flow to the CFCUs from 2550 gpm to 1300 gpm. In addition, the CFCUs start at 60-seconds in the current analyses, whereas their start is delayed to 100-seconds in the revised analyses.

The basis in support of the above assumptions was provided in Enclosure 2 to the licensee's application dated March 16, 2007, Engineering Evaluation (EE) S-C-CBV-MEE-1982, Revision 0, "Updated Containment Pressure/Temperature Response Analysis With SGFP Trip." The NRC staff's evaluation of the key assumptions supporting the licensee's revised containment analysis is included in Safety Evaluation (SE) Sections 3.1.1 through 3.1.3.

3.1.1 SGFP Trip Response and Pump Coastdown

The EE states that the SGFP trip response and coastdown assumed for the Salem analyses is comprised of multiple component delays. The component delays are defined based on a known combination of equipment capability, industry accepted values as approved by the NRC through various license amendments, and evaluation of equivalent assumptions from other plants with similar equipment. The evaluation referenced Diablo Canyon (5-second SGFP trip delay) and Indian Point Unit 3 (7-second SGFP trip delay). The estimated component delay for a Salem SGFP trip is 4-seconds with an additional margin of 3 seconds added for a final assumed 7-second trip delay. The 4-second component delays consist of 2 seconds for SI signal process delay, 1 second for solenoid trip valve/slave relay delay, and 1 second for turbine steam supply stop valves close stroke. The 2-second SI signal process delay was defined previously for Salem and was used in a previous containment analysis (Reference 10). The solenoid trip valve and turbine stop valve time delays are based on consideration of equivalent response requirements for turbine overspeed trips and the fact that the valves are designed for fast closure stop.

The SGFP coastdown time is dependent on the momentum of the rotating elements of both the turbine and the pump, which are influenced by the initial speed, brake horsepower of the turbine, characteristics of the rotating elements, bearing efficiency, piping hydraulic resistance, among others. The EE stated that other reference plants that have credited SGFP coastdown in their MSLB containment analysis have used a range of values between five and 10 seconds. None of the plants have included coastdown values in their TSs. Indian Point Unit 3 assumed

a 10 second coastdown and Diablo Canyon assumed a 5 second coastdown. The EE referenced a license amendment request for Diablo Canyon (Reference 11) to change their TSs, which assumed 5 seconds for SGFP trip and 5 seconds for pump coastdown. The NRC safety evaluation (Reference 12) associated with the amendment request concluded that the response time for the SGFP turbine trip was reasonable and satisfies the assumptions credited in the feedwater analysis. The licensee stated that detailed comparisons were made to Diablo Canyon due to the similarity of the plants and SGFP/turbine drive characteristics. In response to an NRC request for additional information (RAI), the licensee, in a letter dated September 14, 2007 (Reference 3), discussed the sensitivity of the Salem containment analysis if the SGFP takes longer than the assumed 7 second coastdown. The licensee stated than an increase in coastdown time from 7 seconds to 8 seconds has a minimal impact on the integrated mass flow and containment peak pressure. Based on the similarities between the Salem and Diablo Canyon SGFP turbines, the fact that the Salem analysis assumed a higher coastdown time than Diablo Canyon (7 seconds versus 5 seconds) and the licensee's sensitivity analysis indicating a minimal impact on the containment peak pressure if coastdown time were to be even slightly higher than assumed, the NRC staff finds that the SGFP trip and coastdown times used in the revised analysis are reasonable.

3.1.2 FIV Closure Time

In WCAP-16503-NP, the FIV closure time of 32 seconds (includes 2-second SI time delay) is the same as the current analysis, but with a linear reduction in MFW flowrate over the last 10 seconds of the 30-second stroke time versus the last 20 seconds of the stroke time in the current analysis. The EE stated that the design basis stroke time for the valve is 30 seconds as defined in Salem UFSAR Section 15.4.8.2.2. The revised analysis assumes a full-open valve resistance coefficient for the first 20 seconds of the stroke, even though the valve will have completed about 66 percent of the closing stroke. Thus, the valve will be approximately 33 percent open when the model begins the linear decrease in flow. The licensee stated that, in general, the hydraulic resistance of gate valves starts to increase in a linear fashion when the valve flow area is less than 50 percent, except for the final 5 percent, when flow choking may result at the gate valve due to the high pressure drops. Therefore, the licensee concluded that assuming full flow for the first two-thirds of the closing stroke and linearly decreasing flow over the final one-third stroke is appropriate as it delays the reduction in flow and ignores the flow limitation when choking occurs. With respect to the stroke time, the licensee stated that actual tests have indicated a stroke time of approximately 26 seconds and provided recent stroke time test results at Salem Unit 2 in support of this. Based on a review of the information provided, the NRC staff finds that a linear reduction in MFW flowrate over the last 10 seconds of the FIV is reasonable.

3.1.3 CFCU and Delayed Start

The minimum required SW flowrate to the CFCUs will be reduced from 2550 gpm to 1300 gpm. The minimum flow requirement of 1300 gpm consists of a minimum delivered flowrate of 1250 gpm to the CFCU cooling coils and 50 gpm to the CFCU motor cooler. In response to an NRC staff RAI, the licensee clarified the minimum flow requirements in a letter dated August 30, 2007 (Reference 2). The total minimum required heat removal duty used in the revised analysis is 31,325 British thermal units/second (BTU/sec) at a containment temperature of 271 °F (Table 6.1-3 of Reference 7), which equates to 37.6 million BTU/hour per CFCU. Per

a licensee calculation (S-C-CBV-MDC-1637, Revision 3),the minimum required SW flow to the CFCU to obtain this heat load is 930 gpm based on an SW temperature of 93 °F and a thermal fouling resistance of 0.0035 hr -°F-ft^2/BTU. The assumed thermal fouling resistance of 0.0035 hr-°F-ft^2/BTU is greater than the current CFCU design thermal fouling resistance of 0.0030 hr-°F-ft^2/BTU. In addition, the assumed SW supply temperature of 93 °F is conservative, as the current maximum ultimate heat sink temperature is 90 °F. However, a higher flow of 1250 gpm is determined to be necessary to maintain water solid conditions in the CFCU and to ensure fluid conditions at key points in the CFCU return flow path remain above saturation pressure. The licensee stated that an SW system flowpath design modification will be performed to eliminate the modulating control valves and install several fixed diameter orifices for additional hydraulic resistance in order to provide significant flow margin relative to the new TS minimum required flow of 1300 gpm.

The revised analysis in WCAP-16503-NP assumed the start of the CFCUs at 100 seconds. The licensee explained that the current configuration ensures, even with the delays imposed by a loss-of-offsite power (LOOP), the CFCU cooling will be initiated within 60 seconds. However, Salem implemented a design change to meet GL 96-06. This change added head tanks connected to the SW lines using a nitrogen gas cover to ensure the system remained pressurized. This could potentially result in 10 percent degraded heat removal (due to gas entrainment) until the gas is purged out of the CFCUs by the SW flow in the accident mode. The primary concern for CFCUs cooled by an open SW system is column separation when the SW pumps trip and waterhammer following restart of the pumps. To bound the degraded heat transfer, the revised analysis assumes no CFCU heat removal during the first 40 seconds of CFCU SW flow, resulting in a total of 100 seconds delay in crediting the CFCU cooling.

The NRC staff finds that the licensee has appropriately incorporated the reduced heat removal capability of the CFCUs in the revised containment analysis. The credit taken for CFCU cooling is lower than their capability at the new SW flow being implemented. The NRC staff also finds that the 100 second delay in crediting the CFCU cooling is reasonable and conservative.

3.1.4 Crediting Non-Safety Related Equipment

The EE also addressed the acceptability of crediting non-safety related equipment such as the FIV, SGFP trip function, and the SGFP coastdown characteristics in the mitigation of mass addition to a faulted SG during an MSLB event. The licensee stated that NRC's previous acceptance of crediting non-safety related equipment as a response to a single active failure is documented in NUREG-0800, SRP Section 6.2.1.4 - "Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures." The licensee also stated that NRC's position regarding the crediting of non-safety related equipment for the specific case of a single active failure of the feedwater or MSIVs was also addressed by NUREG-0138 and NUREG-0933. In response to an NRC staff RAI, the licensee, in a letter dated November 20, 2007 (Reference 4), provided clarification regarding the extent of the safety-related and non-safety related equipment in the feedwater line and their seismic classification. PSEG stated that the FRVs (main and bypass) and the FIV, and the valve operators were originally procured as non-safety related consistent with the technical guidance from the Nuclear Steam Supply System (NSSS) vendor. However, the FRVs and the FIV have safety-related performance requirements, and receive dual, independent, safety grade trip close signals. In support of this, the licensee quoted UFSAR

Section 15.4.8.2.2, which states, in part:

> The feed water regulating valves (main and bypass) and main feed water isolation valves, which are relied upon to terminate main feed flow to steam generators, are exempt from seismic requirements (thus classified as Seismic Category 3). However, each valve has safety-related performance requirements, and as such receives dual, independent, safety grade, trip close signals from the protection system following a steam line rupture event. The feed water regulating valves are air-operated, fail close design, where as the feed water isolation valves are motor operated. Since the assumed pipe break occurs inside containment in a Seismic Category I pipe, the steam line rupture is not assumed to be initiated by a seismic event. There is no requirement to assume a coincident seismic event with the hypothetical pipe rupture. Thus a seismic classification for the main feed water regulating and isolating valves is not necessary to ensure closure following a steam line break inside containment. Also, since the feed water isolation valves are only credited in the event of a single failure of the regulating valves to close, additional failure of these valves does not need to be considered.

The valves are included in the Salem inservice test (IST) program and are stroke time tested per TS 4.0.5. The FIV motor-operator is maintained and tested in accordance with the GL 89-10, “Motor Operator Valve Program.” As acknowledged in the UFSAR, the safety-related performance requirements for the subject valves are in the current design basis. NUREG-0800 SRP Section 6.2.1.4 - “Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures” states in part that “for the assumed failure of a safety grade steam or feedwater line isolation valves, operation of non-safety grade equipment may be relied upon as backup to the safety grade equipment.” The NRC position regarding the crediting of non-safety related equipment for the specific case of a single active failure of the feedwater or MSIVs was also supported by NUREG-0138 and NUREG-0933. Therefore, the NRC staff finds it acceptable to take credit for non-safety grade equipment in response to single failure as was done in the revised containment analysis for Salem.

3.2 Containment Analysis Results

3.2.1 MSLB

For Unit 1 with Model F SGs, a 1.4 ft^2 double ended rupture (DER) initiated at 100 percent power with an FRV failure results in a peak containment pressure of 41 pounds per square inch gauge (psig) at 243 seconds. For Unit 2 with the current Model 51 SGs, a 1.4 ft^2 DER initiated at 30 percent power with a containment safeguards failure (CSF) results in a peak containment pressure of 42.8 psig at 602 seconds. For Unit 2 with the RSGs, a 1.4 ft^2 DER initiated at 30 percent power with a CSF resulted in a peak containment pressure of 45.6 psig.

The peak containment temperature for both Unit 1 and Unit 2 (with current SGs) is 348.2 °F and it results from a split rupture initiated at 30 percent power with an MSIV failure. For Unit 2 with the RSGs, the peak containment temperature of 349.6 °F resulted from a 0.88 ft^2 split rupture initiated at 30 percent power with an MSIV failure.

3.2.2 LOCA

The maximum calculated containment pressure and temperature for Unit 1 are 40.9 psig and 262.1 °F, respectively. For Unit 2 with the current SGs, the corresponding results are 42.4 psig and 264.4 °F. For Unit 2 with the RSGs, the maximum containment pressure and temperature are 43.5 psig and 265.9 °F, respectively.

It is evident from the above results that Unit 2 would become the most limiting plant and configuration with respect to containment pressure and temperature for both MSLB and LOCA, once the RSGs are in place. The results also show that for an MSLB, CSF has replaced the FRV failure as the governing single failure in determining the peak containment pressure (45.6 psig) and the peak containment temperature (349.6 °F).

3.2.3 Containment Liner Temperature

Section 6.1 of the EE addressed the impact of the results of the revised containment analysis on the containment liner and liner anchors. In response to an NRC staff RAI, the licensee, in a letter dated September 14, 2007 (Reference 3), clarified a discrepancy in Section 6.1 of the EE. The licensee acknowledged that the peak containment temperature for an MSLB in the current analysis of record is 351 °F. Since the peak temperature of the current analysis bounds the revised analysis in WCAP-16503-NP (Reference 7), there is no impact on the containment liner and liner anchors. The licensee indicated that the EE will be revised to provide clarity.

3.3 Conclusions Regarding Containment Analysis

Based on its review as documented in SE Sections 3.1 and 3.2, the NRC staff concludes that the proposed change will continue to meet the requirements of GDCs 16, 38, and 50, and Appendix J to Part 50, as follows:

The Salem Units 1 and 2 containment design limits are defined in TS 5.2.2. Per these requirements, the reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig and containment air temperatures of up to 351.3 °F. The revised analysis shows that the maximum calculated containment pressure and the maximum containment temperature remain below the containment design limits. GDC 16 is satisfied since the proposed change would not result in pressure and temperatures exceeding the containment design limits and since surveillance testing will continue to demonstrate that containment is “essentially leak-tight.”

GDC 38 and 50 are satisfied since the revised containment analysis, which is based on the lower CFCU heat removal rates, does not result in pressures and temperatures exceeding the containment design limits. The containment response analysis for LOCA assumes a LOOP and a postulated failure of an entire train of safeguards equipment. As stated in WCAP-16503-NP (Reference 7), this single failure assumption is conservative because both the Salem units have three emergency diesel generators and the loss of any one single generator would be less limiting than the loss of one complete train of safeguards equipment. In addition, the analysis has shown that the containment pressure for a design-basis LOCA was reduced to less than 50% of the peak calculated pressure within 24 hours after the postulated accident, which is consistent with the guidance in SRP Section 6.2.1.1.A. The MSLB analysis

was based on the most severe single active failure in containment heat removal systems (loss of a train of containment safeguards equipment) or the loss of secondary system isolation provisions (e.g., MSIV failure, FRV failure). The licensee has identified a spectrum of pipe breaks resulting in the highest containment pressure and temperature, pipe break locations and reactor power levels and analyzed the containment for such breaks. The results indicate that pressures and temperatures for both LOCA and MSLB remain below the containment design limits. Therefore, the NRC staff concludes that the proposed change satisfies the requirements of GDC 38, GDC 50.

Section 6.2 of the UFSAR indicates that the Type A test pressure is based on the containment design pressure of 47.0 psig, not the calculated LOCA pressure. Since the design pressure bounds the revised peak containment pressure in WCAP-16503-NP, the Appendix J integrated leak rate testing requirements are not impacted.

3.4 Radiological Dose Consequences

The NRC staff reviewed the licensee's assessments, as they relate to the radiological consequences of DBA analyses, in support of the proposed amendment. The licensee determined, and the staff agrees, that the only DBA analysis which is affected by the new containment integrity analysis is the design-basis LOCA. This is because, of the Salem DBAs, only the postulated LOCA and its assumed resulting plant response is directly affected by changes to the actual post-accident containment pressure, temperature, and ECCS and spray initiation times.

Information regarding the effect of the proposed changes on the design-basis LOCA dose consequence analysis was provided by the licensee in Section 6.3 of Enclosure 2 to the application dated March 16, 2007 (Reference 1). As discussed in SE Section 2.0, the current Salem DBA radiological dose consequence analysis for Salem is based on full-scope implementation of an AST. The following sections address the impact of the licensee's proposed changes on the two affected activity release paths associated with the current DBA analysis.

3.4.1 Containment Leakage

Peak post-LOCA containment pressure is one of the containment response parameters that are affected by the licensee's proposed changes. Currently at Salem, the iodine removal effectiveness of containment sprays is calculated based on an assumed peak containment pressure of 47.0 psig. As discussed above in SE Section 3.3, this value is the containment design limit pressure as shown in TS 5.2.2. The licensee states that the new WCAP-16503-NP containment integrity analysis methodology calculates a new post-LOCA peak containment pressure of 43.5 psig. Mass transfer and spray flow rate both increase with a decrease in containment pressure and would result in more iodine activity being removed by the sprays in containment following the postulated design-basis LOCA. Therefore, the iodine removal effectiveness of the containment sprays assumed in the existing AST analysis remains bounding, from a radiological consequences perspective, in relation to the proposed change.

3.4.2 Auxiliary Building Leakage

As discussed in the NRC staff's SE for the Salem AST amendment dated February 17, 2006, after the reactor water storage tank (RWST) is drained, the ECCS, which is partially located in the Auxiliary Building, is aligned to take suction from the containment sump, such that the contaminated sump water is circulated in the ECCS. During this timeframe, ECCS leakage outside containment provides a path for the release of radionuclides to the environment. The current AST analysis assumes that the release of radionuclides due to ECCS leakage begins 20 minutes into the event. Twenty minutes is the earliest possible timeframe to drain the RWST and initiate realignment of the ECCS suction to the containment sump. In the new WCAP-16503-NP analysis, the licensee determined that containment sump recirculation through the ECCS is initiated no sooner than 29.1 minutes after the initiation of the LOCA. Therefore, Salem's current design basis bounds the effect of the proposed changes with respect to the radiological consequences due to Auxiliary Building leakage following a LOCA.

3.4.3 Radiological Dose Consequences Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed licensing basis changes on the postulated design-basis LOCA. The staff finds that the current design-basis LOCA analysis will bound any impacts of the proposed changes. As a result, the staff concludes that Salem Units 1 and 2 would continue to meet the applicable dose acceptance criteria, as identified in Section 2.0 of this SE, following implementation of the proposed amendment. The staff further finds reasonable assurance that Salem Units 1 and 2, as modified by this proposed amendment, would continue to provide sufficient safety margins, with adequate defense-in-depth, to mitigate unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological dose consequences of DBAs.

3.5 Environmental Qualification (EQ)

The qualification of all EQ equipment inside and outside containment is necessary to ensure that safety-related equipment exposed to the harsh environment of a design-basis event, such as a LOCA or a MSLB, can reliably perform its design function. The EQ analyses use composite curves that envelope the estimated temperature, pressure and radiation environments during a design-basis event.

The current configuration of Salem Units 1 and 2 includes the original Westinghouse Model F and Model 51 SGs, respectively. During the spring 2008 refueling outage, the Salem Unit 2 Model 51 SGs are scheduled to be replaced with Framatome ANP Model 61/19T replacement SGs (RSGs). As discussed in Section 6.5 of Enclosure 2 of the licensee's application dated March 16, 2007 (Reference 1), the bounding temperatures and pressures used for evaluating the impact of the proposed amendment on the EQ of safety-related equipment in containment are from the Unit 2 RSGs containment analysis shown in Appendix A to WCAP-16503-NP (Reference 7). In response to an NRC staff RAI, the licensee, in its letter dated September 14, 2007 (Reference 3), confirmed that the Salem Unit 1 temperature and pressure profiles are bounded by the Salem Unit 2 RSG temperature and pressure profiles.

As discussed in its letter dated September 14, 2007, PSEG uses a software application called EQPro that electronically administers EQ data and provides a means for information access, retrieval, and update. The EQPro software includes automated functions for performing EQ calculations for qualified life and post-accident operability. EQPro was developed in accordance with a quality assurance program, including validation and verification. Use of EQPro at Salem Units 1 and 2 includes quality control requirements specified by applicable PSEG procedures.

3.5.1 Containment Temperature Impact

Figures 6.5-1, 6.5-2, and 6.5-3 in Enclosure 2 to the licensee's application dated March 16, 2007, compare the Unit 2 RSG design-basis event containment temperatures (based on the revised containment analysis in WCAP-16503-NP) to the current EQ temperature envelope (i.e., analysis of record (AOR)). The three figures (hot leg break LOCA, cold leg break LOCA, and MSLB) also show the EQPro input curves. As discussed in the licensee's letter dated September 14, 2007, the EQPro input curves were used to develop the bounding design-basis event temperature profile for the qualification of the EQ equipment.

As discussed in Section 6.5 of Enclosure 2 to the application dated March 16, 2007, the containment temperatures in the revised containment analysis exceeded the existing AOR in portions of each of the three cases. However, at no time was the containment design temperature of 351 °F exceeded. The licensee stated, in its letter dated September 14, 2007, that in all areas where the AOR was exceeded, the EQ composite temperature profile was increased and the EQ equipment reanalyzed and qualified to the revised conditions. The licensee stated that the EQ equipment was qualified by direct comparison of equipment test data to the new EQ temperature profile and by utilizing the Arrhenius aging calculation. Post accident operability was evaluated by extrapolating vendor EQ equipment test data against the new EQ temperature profiles. The licensee also stated that the results indicated that all EQ equipment inside containment has greater than 10 percent test margin and remained qualified to the worst case pipe break cases analyzed for the RSGs.

Based on review of the information provided by the licensee as described above, the NRC staff finds that the licensee has adequately addressed the impact of the proposed amendment on containment temperature and the qualification of EQ equipment.

3.5.2 Containment Pressure Impact

As discussed in Section 6.5 of Enclosure 2 to the application dated March 16, 2007, the licensee stated that the containment pressure increase attributed due to the proposed amendment was bounded by the current design-basis composite curve of 47 psig and will not impact the previously analyzed EQ equipment.

As discussed in SE Section 3.2, the revised containment analysis showed that the highest containment pressure would be 45.6 psig (MSLB for Salem Unit 2 with RSGs). Since this pressure is less than 47.0 psig value used in the existing EQ analysis, the NRC staff agrees with the licensee's conclusion that the containment pressure increase attributed due to the proposed amendment would have no impact on the qualification of EQ equipment.

3.5.3 Containment Radiation Impact

In response to an NRC staff RAI, the licensee, in its letter dated September 14, 2007, confirmed that the radiation environment did not change due to the proposed amendment. Therefore, the NRC staff finds that the proposed amendment does not impact the qualification of EQ equipment from a radiation perspective.

3.5.4 Outside Containment Impact

The NRC staff requested the licensee to provide an evaluation of any impact on the outside containment EQ equipment due to the proposed amendment. The licensee in its letter dated September 14, 2007, stated that it has evaluated the thermal response of the safety-related equipment outside containment in the outboard main steam penetration access area. The evaluation compares the EQ test data to the data resulting from the RSG project accident conditions. Based on the results of this evaluation the licensee concluded that all EQ equipment in this area remains qualified to the new transients resulting due to this proposed amendment.

Based on review of the information provided by the licensee, the NRC staff finds that the licensee has adequately addressed the impact of the proposed amendment on EQ equipment located outside of containment.

3.5.5 EQ Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed amendment on EQ equipment as discussed above. No adverse impacts on the EQ equipment were identified as a result of the proposed amendment. In addition, no EQ equipment has been identified to be requalified or replaced due to the proposed amendment. The NRC staff finds that the licensee has adequately addressed: (1) the effects of the proposed amendment on the environmental conditions; and (2) the impact on the EQ equipment due to the proposed amendment. Based on these findings, the NRC staff concludes that there is reasonable assurance that the EQ equipment would continue to perform their intended functions following a LOCA or MSLB for the environmental conditions evaluated for the proposed amendment. Therefore, the NRC staff concludes that the proposed amendment is acceptable with respect to the requirements in 10 CFR 50.49.

3.6 <u>CFCU Performance</u>

As discussed in SE Section 3.0, the licensee's revised containment analysis credits a reduced heat removal capability of the CFCUs that allows a reduction of required SW cooling flow to the CFCUs. Currently, the TS SRs 4.6.2.3.b.3 and 4.6.2.3.c.2 require a minimum cooling water flow rate of 2550 gpm to the CFCUs. The proposed change would reduce the minimum flow rate to 1300 gpm to reflect the revised containment analysis.

The reduction of the required SW flow to the CFCUs would reduce their heat removal capability for the containment. The minimum SW flow rate requirement in the licensee's analyses are based on clean-tube flow rate. In its application dated March 16, 2007, thermal performance degradation caused by tube fouling was not addressed. In response to an NRC staff RAI, the licensee, in its letter dated August 30, 2007, explained: (1) how the accident analysis was

performed with consideration of tube fouling; and (2) what safety margin was assumed in the analysis for tube fouling. The licensee stated that the CFCU performance curve used in the revised containment analysis (i.e., WCAP-16503-NP) was developed with sufficient conservatism to address the effects of fouling. The total minimum required heat load at 271 °F from Table 6.1-3 of WCAP-16503-NP is 31,325 BTU/sec, which equates to 37.6 MBTU/hr per CFCU. The minimum SW flow to the CFCUs to obtain this heat load, which is addressed in the licensee's calculation, is 930 gpm, based on an SW temperature of 93 °F and a thermal fouling of 0.0035 hr-°F-ft^2/BTU. The assumed thermal fouling resistance of 0.0035 hr-°F-ft^2/BTU is greater than the current CFCU design thermal fouling resistance of 0.0030 hr-°F-ft^2/BTU.

The licensee stated that system design with fixed resistance modifications has margin to accommodate degradation of CFCU thermal performance from thermal fouling, bio-fouling, or silt accumulation. This margin results from the following considerations:

1. The assumed SW supply temperature of 93 °F is conservative, as the current system design value is 90 °F.

2. The minimum required SW flow (1300 gpm total, including a 50 gpm allowance for the motor cooler) with the fixed resistance modifications is higher than necessary to meet the heat duty requirement in order to ensure no two-phase flow in the system (1250 gpm versus 930 gpm).

Based on an SW flow of 1250 gpm and a temperature of 90 °F, the maximum thermal fouling to obtain the minimum required heat load is about 0.0055 hr-°F-ft^2/BTU, which is 83 percent greater than the current design value.

The NRC staff has reviewed the licensee's response and finds that the revised containment analysis has considered the effects of thermal fouling. The analysis showed that the assumed thermal fouling resistance is greater than current CFCU design thermal fouling resistance. Since the licensee has provided a sufficient basis to prove that the system design has the margin to accommodate degradation of the CFCU thermal performance, the NRC staff concludes that the licensee's response is acceptable.

In considering the effects of water-side surface fouling of fan coolers and residual heat removal (RHR) heat exchangers, the NRC staff asked the licensee to explain how the water-side surface fouling of the fan coolers was treated in the analysis. In its response dated August 30, 2007, the licensee stated that the plant is designed to continuously chlorinate the SW system and has an administrative 7-day limit on having the chlorination system out of service. This requirement is detailed in a chemistry procedure and is related to the licensee's response to GL 89-13. If the system is out of service for more than 7 days, a corrective action program notification is required per the same procedure. The 7-day limit is to allow for maintenance on the system. The licensee stated that chlorination has proven to be highly effective in preventing bio-fouling, as confirmed by thermal performance testing of the component cooling heat exchangers. The licensee stated that the existing GL 89-13 commitments for the plant are not being modified by this proposed amendment and that all of the previously committed actions will continue to be performed by the station.

The licensee also stated that an evaluation was performed to assess the potential silting within the CFCU tubes before and after the fixed hydraulic resistance modification. This evaluation determined the maximum size particle that can be maintained in a symmetrical suspension for a given flow rate and the maximum particle size that will not deposit at the same flow rate. The evaluation found that a flow velocity of 3 ft/sec is sufficient to keep particles up to 500 microns in diameter in a symmetric suspension and particles up to almost 1500 microns in an asymmetric suspension. The large majority of particles in the Delaware River samples at the plant site are under 200 microns and a few are greater than 400 microns. With the fixed hydraulic resistance modifications, the velocity for normal operation is greater than 3 ft/sec, which is sufficient to transport particles larger than what is typically seen in the river samples taken at the plant site.

The licensee concluded that the increased SW flow during normal operations and chlorination of the SW system will be sufficient to ensure that the water side of the CFCU tubes remains free of surface fouling. The NRC staff has reviewed the information provided by the licensee, as described above, and finds that the licensee's conclusion is reasonable.

The cooling water system serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios. However, the heat removal assumptions for the design-basis accident scenarios were based on single-phase flow conditions. In GL 96-06, the NRC requested licensees to evaluate cooling water systems that serve containment coolers to assure that the systems are not vulnerable to water-hammer and two-phase flow conditions. The NRC staff has previously approved the licensee's evaluations and corrective actions to resolve this issue in a letter dated June 2, 2003. The staff asked the licensee to provide additional information on this issue based on the new parameters for the proposed TS changes.

In its response dated August 30, 2007, the licensee stated that the proposed TS change would reduce the minimum CFCU flow rate from 2550 gpm to 1300 gpm. The licensee states that the minimum CFCU flow of 1300 gpm (1250 gpm to the CFCU plus 50 gpm for the motor cooler) is adequate to meet the CFCU heat removal assumptions and prevent flashing in the SW system with maximum heat transfer conditions. The licensee's calculation established the minimum CFCU flow requirements. The calculation considers the need to satisfy both the required accident heat removal capability and the requirement to prevent two-phase flow.

The minimum CFCU flow necessary to ensure the CFCUs can satisfy the heat removal capability credited in WCAP-16503-NP is 930 gpm. This value is based on an SW supply temperature and a thermal fouling factor that are greater than the current design values. The SW temperature of 93 °F provides 3 °F margin relative to the maximum ultimate heat sink (UHS) temperature of 90 °F. The assumed tube-side fouling factor of 0.0035 hr-°F-ft^2/BTU is greater than the value previously used to assess minimum heat transfer with maximum fouling (0.0030 hr-°F-ft^2/BTU). Therefore, the SW flow of 930 gpm is sufficient to satisfy the heat removal requirement of the CFCU with margin for the degraded heat transfer. Accounting for the minimum required flow to the motor (25 gpm), the total CFCU loop flow necessary to meet the CFCU and associated motor cooler heat duty is 955 gpm.

The minimum flow necessary to maintain water solid conditions in the CFCU and associated return piping path is determined by a licensee calculation. It uses a conservative saturation margin analysis to demonstrate that 1250 gpm (1300 gpm total, including a 50 gpm allowance

for the motor cooler) is sufficient to ensure fluid conditions remain above saturation pressure during worst case accident conditions (i.e., system aligned to provide minimum static pressure coupled with maximum heat transfer conditions in the CFCUs with zero thermal fouling and a SW supply temperature of 93 °F). The evaluation uses the maximum containment temperature for a LOCA from WCAP-16503-NP to assess saturation margin at key points in the CFCU return flow path. The system pressures are determined using the benchmarked hydraulic model of the SW system. The CFCU outlet temperature is determined using a benchmarked thermal model of the CFCUs. The results of the saturation margin evaluation show positive saturation at each of the key locations evaluated.

The approach used in the saturation margin evaluation includes significant conservatism due to the difference in CFCU heat removal rate used in the containment evaluation and that used for the CFCU outlet temperature. The maximum containment temperature for a LOCA from WCAP-16503-NP is used to determine the CFCU outlet temperature. The containment temperature profile from WCAP-16503-NP credits an individual CFCU heat removal rate of 37.6 MBTU/hr, whereas the heat removal rate calculated with zero fouling and a CFCU flow of 1250 gpm is estimated at 70.0 MBTU/hr. Such a large increase in heat removal would be expected to reduce containment parameters resulting in the corresponding reduction in SW outlet temperature. The design of the fixed resistance modifications provides significant flow margin relative to the new TS minimum required flow of 1300 gpm.

The licensee also stated that the above discussion focuses on the steady-state response to a LOCA or MSLB with or without a concurrent LOOP event. GL 96-06 required evaluation of the CFCUs for potential water-hammer transient events following LOOP and LOOP/LOCA events. The primary concern for CFCUs cooled by an open SW system is column separation when the SW pumps trip and the subsequent water-hammer via column impact following restart of the SW pumps. The licensee's response to GL 96-06 was to install two 15,000-gallon accumulators on the SW system headers. The existing SW accumulator tanks prevent two-phase flow conditions and potential water-hammer transients following LOOP or LOOP/LOCA. The function of the SW accumulators is to inject water into each SW header during a LOOP to ensure that the SW piping, particularly downsteam of the CFCUs, are maintained in single-phase flow conditions until the SW pumps are restarted from an emergency diesel generator. This SW system design feature ensures that water-hammer cannot occur as two-phase conditions are not allowed to develop. The SW accumulator pressure and water level, which are based on the current modulating flow controls for the system, are conservative for the fixed resistance modifications due to the increase in system resistance. The accumulator tanks will prevent two-phase flow conditions and potential water-hammer transients following a LOOP or LOOP/LOCA. Therefore, the 1300-gpm minimum flow for the CFCU is more than adequate to meet the CFCU heat removal assumptions in WCAP-16503-NP and is sufficient to ensure no two-phase flow for accident conditions.

The licensee performed analyses for the design-basis LOCA and MSLB transients to determine the acceptability of 1300 gpm SW flow rate for the CFCU to perform its heat removal for the containment during accident conditions. Based on review of the analyses, the NRC staff found that the minimum SW flow rate specified in the submittal meets the containment cooling requirements and prevents two-phase flow conditions and water-hammer transients. Therefore, the licensee's previous evaluation and commitment to GL 96-06 is still valid for the new

parameters. Based on this review, the NRC staff concludes that the licensee's response is acceptable.

Based on the above review, the NRC staff found that the licensee has performed safety analyses to support the CFCU margin recovery project. The staff's concerns regarding the impact of cooling water reduction on CFCU performance and heat removal degradation due to fouling were included in the analyses. Also, the licensee has performed thermal analysis for the CFCU to prove that the licensee's previous response to GL 96-06 to prevent two-phase flow and water-hammer for accident conditions remains valid for the new TS parameters. On the basis of this review, the staff concludes that the proposed amendment is acceptable with respect to CFCU performance.

3.7 Overpressurization of Piping Penetrating Containment

GL 96-06 requested that licensees evaluate the potential for overpressure of piping systems that penetrate the containment due heating of trapped fluid. Section 6.8 of Enclosure 2 of the licensee's application dated March 16, 2007, provided the licensee's evaluation regarding the impact of the revised containment temperature and pressure response on analyses that were performed in response to GL 96-06. The submittal indicated that the six Salem Unit 1 penetrations that were potentially affected by the GL 96-06 concern were protected by relief valves. The submittal also indicated that two of the six Salem Unit 2 penetrations were protected by installing overpressure protection. PSEG concluded that these penetrations are not affected by the higher peak LOCA temperature because they are protected by the existing relief valves. The NRC staff agrees with PSEG's conclusion.

PSEG indicated that the evaluation of Salem Unit 2 penetration M25 was based on a LOCA containment temperature of 260 °F which resulted in an average trapped water temperature of 126.9 °F. PSEG also indicated that the maximum peak LOCA temperature of 265.9 °F, per the revised containment analyses shown in WCAP-16503-NP, will result in a small change in the average trapped water temperature in penetration M25. PSEG stated that a scoping calculation found that the change in average trapped water temperature will result in a small (less than 1 psia) change in pressure. PSEG also indicated that there is sufficient design margin in the piping and isolation valves to accommodate a significant increase in pressure. On the basis of PSEG's evaluation, the NRC staff concludes that the design of penetration M25 has sufficient margin to accommodate the change in peak LOCA temperature.

PSEG application dated March 16, 2007, also provided an evaluation of Salem Unit 2 penetrations M22A, M27, and M45 for the potential increase in pressure as a result of the higher LOCA containment temperature. However, this evaluation was not consistent with PSEG's May 8, 1998, response to an NRC staff RAI regarding GL 96-06. The RAI response indicated that, for Salem Unit 2 penetrations M22A, M27, and M45, relief valves would be installed to protect these penetrations from thermally-induced pressurization. By letter dated August 30, 2007, PSEG confirmed that relief valves had been installed to protect penetrations M22A, M27, and M45. Therefore, the NRC staff concludes that these penetrations are not affected by the higher peak LOCA temperature.

On the basis of the above evaluation, the NRC staff finds that the previous staff conclusions regarding thermally-induced overpressure of piping and containment penetrations are not impacted by the change in peak LOCA temperature calculated for the proposed amendment.

3.8 Core Cooling

The NRC staff reviewed the proposed amendment with respect to its affect on core cooling. Following a LOCA in a pressurized-water reactor (PWR) plant, the ECCS supplies water to the reactor vessel to reflood, and thereby cool, the reactor core. The core flooding rate is governed by the capability of ECCS water to displace the steam generated in the reactor vessel during the core reflooding period. For PWR plants, core flooding rate depends directly on containment pressure (i.e., the core flooding rate increases with increasing containment pressure). A decrease in containment pressure could reduce the effectiveness of the ECCS.

As discussed in Section 6.2.1 of Enclosure 2 of the licensee's application dated March 16, 2007, the analysis related to the LOCA core response is conservatively biased by minimizing the containment backpressure such that the safety injection effectiveness is minimized (the core becomes the highest resistance flow path). This bias is the result of assuming that the SW temperature supplied to the operating CFCUs is cold (32 °F), and that RWST water being supplied to the containment sprays is cooler than water being provided to the safety injection system. Also, containment spray flow is maximized based on minimum containment pressure (7302 gpm at 10 psig) to conservatively reduce the containment backpressure. The licensee noted that none of these assumed boundary conditions rely on the minimum CFCU heat removal capacity as the primary means of maintaining containment pressure. Based on these considerations, the licensee concluded that any reduction in the accident capability of the CFCUs has no bearing on the LOCA core response. The NRC staff agrees with the licensee's conclusion.

Based on review of the information provided by the licensee, as discussed above, the NRC staff concludes that the proposed amendment will have no adverse impact on core cooling. Therefore, the staff concludes that the proposed amendment is acceptable with respect to the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

3.9 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 through 3.8, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the

amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 17951). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. PSEG letter LR-N07-0030, “License Change Request S06-010, Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System,” dated March 16, 2007.

2. PSEG letter LR-N07-0021, “Response to RAI#1 and RAI#2 on LCR S06-10 (TAC Nos. MD4843 and 4844) Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System,” dated August 30, 2007.

3. PSEG letter LR-N07-0222, “Response to RAI#3 on LCR S06-10 (TAC Nos. MD4843 & 4844) Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System,” dated September 14, 2007.

4. PSEG letter LR-N07-0293, “Response to RAI#4 on LCR S06-10 (TAC Nos. MD4843 and 4844) Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System,” dated November 20, 2007.

5. “Mass and Energy Releases Following a Steam Line Rupture,” WCAP-8822 (Proprietary), WCAP-8860 (Nonproprietary), Land, R.E., September 1976.

6. Letter from Cecil O. Thomas (NRC), “Acceptance for Referencing of Licensing Topical Report WCAP-8821 (P)/8859(NP),” “TRANFLO Steam Generator Code Description,” and WCAP-8822(P)/8860(NP), “Mass and Energy Release Following a Steam Line Rupture,” August 1983.

7. WCAP-16503-NP, Revision 3, “Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit Margin Recovery Project,” February 2007 (Enclosure 1 to PSEG letter LR-N07-0030 dated March 16, 2007).

8. “LOFTRAN Code description,” WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), Burnett, T.W.T., et al., April 1984.

9. “Containment Pressure Analysis Code (COCO),” WCAP-8327, July, 1974 (Proprietary), WCAP-836, July, 1974 (Nonproprietary).

10. Westinghouse Project Letter, PSEBO-97-022, “Safety Evaluation for Revised Fan Cooler Delay Time (SECL-96-178, Revision 2)”, dated September 2, 1997.

11. Pacific Gas and Electric Company Letter DCL-98-109, License Amendment Request 98-05, dated August 10, 1998.

12. NRC Letter to PGE, “Diablo Canyon Nuclear Power Plant, Units 1 and 2 - Issuance of Amendment Re: Main Feedwater System (TAC Nos. MA3407 and MA3408”, dated February 22, 2000.

13. PSEG letter LR-N08-0009, “Supplement to LCR S06-010 (TAC Nos. MD4843 & MD4844) - Mark-up of UFSAR Tables, Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System,” dated January 16, 2008.

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Date: February 27, 2008