March 5, 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 NO. 2, ISSUANCE OF AMENDMENT RE: REFUELING OPERATIONS - DECAY TIME (TAC NO. MD7027)

Dear Mr. Levis:

The Commission has issued the enclosed Amendment No. 271 to Facility Operating License No. DPR-75 for Salem Nuclear Generating Station (Salem), Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 17, 2007, as supplemented by letter dated January 11, 2008. The amendment allows a one-time revision to the requirements for fuel decay time prior to commencing movement of irradiated fuel in the reactor. Specifically, the proposed amendment revises TS 3/4.9.3 to allow fuel movement to commence at 86 hours after the reactor is subcritical. The proposed change is only applicable to Salem Unit 2 refueling outage 2R16 which is scheduled to commence on March 11, 2008.

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 No. 50-311

Enclosures:
1. Amendment No. 271 to License No. DPR-75
2. Safety Evaluation

cc w/encls: See next page

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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. 271
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 October 17, 2007, as supplemented by letter dated January 11, 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. 271 , 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 within 7 days.

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: March 5, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 271

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of Facility Operating License No. DPR-75 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
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 9-3	3/4 9-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 271 TO FACILITY OPERATING LICENSE NO. DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated October 17, 2007, as supplemented by letter dated January 11, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML073470363 and ML080230549, respectively), PSEG Nuclear LLC (PSEG or the licensee) submitted an amendment request for changes to the Salem Nuclear Generating Station (Salem), Unit No. 2 Technical Specifications (TSs). The proposed amendment would allow a one-time revision to the requirements for fuel decay time prior to commencing movement of irradiated fuel in the reactor pressure vessel (RPV). Currently, TS 3/4.9.3, "Decay Time" requires that: (a) the reactor has been subcritical for at least 100 hours[1] prior to movement of irradiated fuel in the RPV between October 15th and May 15th; and (b) the reactor has been subcritical for at least 168 hours prior to movement of irradiated fuel in the RPV between May 16th and October 14th. The calendar approach is based on average river water temperature which is cooler in the fall through spring months. The proposed amendment would revise TS 3/4.9.3 to allow fuel movement to commence at 86 hours after the reactor is subcritical. The proposed change would only be applicable to Salem Unit 2 refueling outage 2R16 which is scheduled to commence on March 11, 2008.

The supplement dated January 11, 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 December 4, 2007 (72 FR 68218).

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1 The current TS decay time requirement of 100 hours is applicable through the year 2010. After 2010, the 168 hour decay time requirement would be applicable for refueling outages occurring between October 15th and May15th.

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 October 17, 2007. The regulatory requirements, criteria, and guidance which the NRC staff applied in its review are discussed below.

As discussed in Section 3.1 of the Salem Updated Final Safety Analysis Report (UFSAR), 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. Section 3.1.3 of the UFSAR also states that the Salem Unit 1 and 2 design also conforms with the intent of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," dated July 7, 1971, except for several specific exceptions as noted in that UFSAR section. The following GDCs in Appendix A to 10 CFR Part 50 are applicable to the proposed amendment:

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

- GDC 61, "Fuel storage and handling radioactivity control," insofar as it specifies that fuel storage systems shall be designed with: (1) a residual heat removal system capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and (2) the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

UFSAR Section 15.4.6 describes the current Salem licensing basis analysis for a fuel handling accident (FHA). The current analysis uses a fuel decay time of 96 hours which bounds the current minimum decay time of 100 hours required by TS 3.9.3. The licensee's current analysis for an FHA is based on selective use of an alternative source term (AST) in accordance with 10 CFR 50.67, "Accident source term," and the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The selective implementation of the AST methodology for an FHA was approved by the NRC for Salem Units 1 and 2 in Amendment Nos. 251 and 232 on October 10, 2002 (ADAMS Accession No. ML022770181). Full scope implementation of the AST methodology for Salem Units 1 and 2 was approved by the NRC in Amendments 271 and 252 on February 17, 2006 (ADAMS Accession No. ML060040322).

To support the proposed one-time TS 3.9.3 change to use a fuel decay time of 86 hours, the licensee re-analyzed the radiological consequences of a design-basis FHA using a bounding minimum decay time of 24 hours. The NRC staff evaluated the impact of the proposed change in decay time on the previously analyzed radiological consequences for an FHA and the acceptability of the revised analysis results. Consistent with 10 CFR 50.67, GDC 19, and Regulatory Position C.4.4 in RG 1.183, the applicable acceptance criteria for an FHA are 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE) in the control room, 6.3 rem TEDE at the exclusion area boundary (EAB), and 6.3 rem TEDE at the outer boundary of the low population zone (LPZ). Except where the licensee proposed a suitable alternative, the

NRC staff used the guidance provided in applicable sections of RG 1.183 and NUREG-0800, Standard Review Plan (SRP), Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms” and SRP Section 6.4, “Control Room Habitability System,” in performing this review.

Section 9.1.3 of the UFSAR describes that the spent fuel cooling system provides cooling to remove residual heat from the fuel stored in the spent fuel pool (SFP). The system is designed with reliability and redundancy to assure continued heat removal under normal, abnormal, and accident conditions. Section 9.1.3 of the UFSAR also describes SFP temperature limits of 149 °F when both heat exchangers are available and 180 °F when only one spent fuel cooling system heat exchanger is available to cool both pools. The SFP structure, including the liner, has been evaluated by the licensee for performance of its design function at temperatures of 180 °F. These descriptions provide criteria applicable to the NRC staff’s evaluation of the proposed amendment with respect to SFP cooling.

3.0 TECHNICAL EVALUATION

Before movement of any irradiated fuel from the RPV to the SFP, a period of in-vessel decay is required. This is because of post design-basis accident radiation dose consequence limitations and fuel pool cooling capacity requirements. With regard to pool cooling, decay heat from irradiated fuel assemblies decreases as the fission products that dominate heat generation decay. Also, due to radioactive decay, the amount of radioactivity available for release from the fuel is reduced over time. Therefore, the longer the fuel elements are allowed to decay within the RPV before being moved, the less heat is transferred to the SFP, and the lower the dose consequence of a radioactive release resulting from an FHA.

The current Salem licensing basis includes an 100-hour decay time requirement for outage periods falling between October 15th and May 15th and an 168-hour decay time requirement for outage periods falling between May 16th and October 14th. The 100-hour decay time requirement was included in TS 3.9.3 via Amendments 251 and 232 which were issued by the NRC on October 10, 2002. These amendments represented a change from the previous year-round decay time requirement of 168 hours. In a previous, and now withdrawn, amendment request dated August 4, 2006, the licensee proposed a change that would have allowed the movement of irradiated fuel in the RPV to commence at 24 hours after shutdown or at a decay time calculated using the licensee’s integrated decay heat management (IDHM) program, whichever is later. That change would have required a pre-outage assessment of the SFP heat loads and heat-up rates to assure available SFP cooling capability prior to offloading fuel. For this current amendment request, the licensee proposes to revise only the Salem Unit 2 licensing basis for the upcoming spring 2008 refueling outage (2R16) to require an 86-hour fuel decay time before movement of irradiated fuel, by revising TS 3/4.9.3 “Decay Time.” This amendment is only be applicable to the aforementioned upcoming refueling outage, 2R16.

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

3.1 FHA Radiological Consequences
3.2 SFP Cooling
3.3 Technical Evaluation Conclusion

3.1 FHA Radiological Consequences

The NRC staff reviewed the information provided by the licensee in support of the proposed license amendment, related to the radiological consequences of a design-basis FHA, as described below. Section 4.2 of Attachment 1 to the licensee's application dated October 17, 2007, "Fuel Handling Accident Radiological Consequences," provides a summary of the licensee's analysis. Attachment 5 to the application provides the licensee's detailed FHA radiological consequences evaluation (Calculation S-C-ZZ-MDC-1920).

3.1.1 FHA Analysis

As discussed above in Safety Evaluation (SE) Section 2.0, to support the proposed one-time TS 3.9.3 change to use a fuel decay time of 86 hours, the licensee re-analyzed the radiological consequences of a design-basis FHA using a bounding minimum decay time of 24 hours. The licensee's calculation supporting the revised FHA analysis was performed using the AST guidance of RG 1.183, and the TEDE dose criteria.

The licensee calculated the TEDE to the maximally exposed individual offsite and in the control room for the design-basis FHA occurring in containment and the fuel handling building (FHB), for decay times of 24, 48, 60, 72, and 96 hours. The licensee's analysis assumes no containment closure during fuel movement with the activity released to the environment through either the open containment equipment hatch or the plant vent.

The licensee's calculation re-evaluated the dose for the design-basis FHA radiological consequences in the containment building and in the FHB using AST methodology. The inventory of fission products in the reactor core is a function of the reactor power, the duration of the at-power operation, and the time after shutdown prior to movement of spent fuel. The current licensed maximum thermal power level for Salem Unit 2 is 3459 megawatts thermal (MWt). As discussed in UFSAR Section 15.4.6, the current FHA analysis assumes that the reactor had been operating at 3600 MWt prior to shutdown. The licensee's revised analysis used a power level of 3632 MWt, or 105 percent of the current licensed maximum thermal power level, to provide margin in the event of a future power uprate. The NRC staff finds that the power level assumption is conservative and acceptable.

The licensee's analysis used a radial peaking factor of 1.7 to account for differences in the power distribution throughout the core. This assumption is the same as the current licensing basis for Salem, as described in UFSAR Section 15.4.6, and is acceptable.

The licensee assumed fuel gap activity fractions larger than those previously approved for Salem. The gap fission product fraction values in Table 3 of RG 1.183 are multiplied by a factor of 2, as shown in Section 5.3.1.3 of Attachment 5 to the licensee's application dated October 17, 2007. The licensee states that they currently limit the Salem peak rod average power to a maximum linear heat generation rate of 6.3 kilowatts/foot (kW/ft). For the FHA, the licensee doubled the assumed gap fractions to establish a conservative basis for fuel assemblies that may, in future cycles, exceed this maximum linear heat generation rate. The NRC staff finds the doubling of the RG 1.183, Table 3, fission product activity gap fractions to be a conservative assumption for fuel that exceeds the RG 1.183, footnote 11 limitation on fuel burnup and linear heat generation rate.

FHA Occurring in the Containment Building

The licensee assumed that a single fuel assembly is dropped in the reactor cavity pool resulting in fuel cladding damage to all the fuel rods in the dropped assembly. Transport and release of radioactivity to the environment for the design-basis FHA is modeled assuming no containment closure during fuel movement (e.g. personnel air locks and other containment penetrations open). The activity is assumed to be released to the environment through the opened containment equipment hatch or plant vent over a 2-hour time period.

The licensee considered one containment equipment hatch, two personnel air locks, and containment piping penetrations through the containment boundary as potential release points of radionuclides to the environment. The containment equipment hatch provides a direct release path to the environment. The personnel air locks and penetrations provide release paths to the environment through the plant vent via piping penetration areas. The licensee used the most limiting atmospheric dispersion factors (χ/Qs) determined from these release paths to bound the dose consequence of a release from any of the openings.

A summary of the licensee's calculated results of an FHA occurring in the containment building is given in Section 8.1 of Attachment 5 to the licensee's application dated October 17, 2007, and is included in Tables 1 and 4 of this SE. The analysis for the 24-hour decay time gives the maximum dose results due to the least amount of radioactive decay. The licensee's analysis, assuming 24 hours of decay time, shows that an FHA occurring in the containment building would result in TEDE values of 1.26 rem TEDE for the EAB and 0.18 rem TEDE for the LPZ. These results are below the regulatory dose acceptance criterion of 6.3 rem TEDE for both the EAB and LPZ, as shown in RG 1.183 and SRP 15.0.1. The licensee's FHA dose results for the control room are discussed below.

FHA Occurring in the Fuel Handling Building

For an FHA occurring in the FHB, the licensee assumed that a single fuel assembly is dropped in the SFP, resulting in fuel cladding damage to all the fuel rods in the dropped assembly. The radionuclides, assumed to be released from the damaged fuel rods, pass through the water in the reactor cavity, or SFP, and enter the FHB atmosphere instantaneously. In the licensee's analysis, the release from the FHB to the environment is assumed to mix in the FHB volume and be released over a 2-hour time period. This 2-hour release period is consistent with the guidance of RG 1.183. The licensee's analysis assumptions, shown in the license amendment request, are the same as in the current licensing basis FHA analysis.

The licensee assumed a combined release through: (1) the plant vent, via one operational FHB exhaust fan, at a rate of 15,300 cubic feet per minute (cfm); (2) through the truck bay roll-up door at a rate of 3883 cfm; and (3) through the gravity damper (modeled as the smoke hatch) at a release rate of 256 cfm.

The licensee performed a parametric study to determine a conservative release model using either a post-FHA release rate based on a 0-2 hour release due to an exhaust fan failure, or a rapid release rate based on one FHB volume per minute. The results of the parametric study, shown in Sections 8.2 and 8.3 of Attachment 5 to the licensee's application dated October 17, 2007, indicate that a release based on the rapid release rate of one FHB volume per minute yields a higher control room dose, and is the limiting event, with regard to the resulting control room dose. The puff release yields a higher control room dose because it results in a larger amount of unfiltered iodine activity entering the control room volume prior to the assumed 1-minute delay in the automatic start of the Control Room Emergency Air Conditioning System (CREACS) outside air inflow filtration. The puff release yields a slightly higher dose (0.01 rem) than the 0-2 hour release for the EAB, while the LPZ dose for both cases is identical.

The licensee's calculated dose results of an FHA occurring in the FHB are included in Tables 2 through 4 of this SE. The licensee's analysis shows that, for an FHA occurring in the FHB, the limiting event would be that of a rapid release assuming a 24-hour decay period, which would result in TEDE dose values of 1.27 rem for the EAB and 0.18 rem TEDE for the LPZ. These results are below the regulatory dose acceptance criterion of 6.3 rem TEDE for both the EAB and LPZ, as shown in RG 1.183 and SRP 15.0.1. The licensee's FHA dose results for the control room are discussed below.

Control Room Habitability

The CREACS operates to ensure continuous occupancy of operational personnel in the control room envelope under emergency conditions. The CREACS supplies cooled high-efficiency particulate air (HEPA) and charcoal filtered air to the control room envelope when actuated by a control room intake high radiation signal. As discussed in Salem UFSAR Section 9.4, CREACS is cross-tied between Salem Units 1 and 2, and both units operate simultaneously in pressurized mode during a design-basis accident.

In addition to CREACS, the Control Room Envelope Pressurization System (CREPS) is designed to ensure that a positive pressure is maintained in the control room envelope for any event with the potential for radioactive release. The positive pressure supplied by CREPS limits control room inleakage and, consequently, dose to the control room occupants.

The control room effective volume used in the calculation of control room habitability is 81,420 cubic feet. The normal control room ventilation intake flow rate prior to isolation is 1320 cfm. The control room will isolate on a control building isolation signal from the control room intake radiation monitor.

The period prior to CREACS initiation and after control building isolation is referred to as the control room neutral condition. During the neutral condition, the control room is isolated, the normal ventilation flow rate of 1320 cfm has terminated, and the CREACS is not operating. During the neutral condition there is no mechanically-induced ventilation of the control room. For the neutral condition, the licensee conservatively has assumed an unfiltered inleakage of 150 cfm for use in control room habitability calculations.

The licensee evaluated the dose to operators in the control room. With the assumed normal outside air flow rate of 1320 cfm, the licensee also assumes that the control room ventilation system automatically realigns, within 1 minute, to an emergency configuration. In this mode, the normal outside air makeup is halted and filtered emergency pressurization is started at a flow rate of 2200 cfm. The flow rate through control room recirculation filters is 5000 cfm.

During the control room positive pressure period, the licensee's analysis assumes an unfiltered inleakage of 150 cfm, which is conservatively based on tracer gas testing results. Once initiated, CREACS operation and the associated control room positive pressure period persists for the duration of the postulated event. The iodine activity removal efficiency attributed to the CREACS, as assumed by the licensee, is 95 percent for aerosol, elemental, and organic chemical forms of iodine. With regard to the aerosol iodine chemical form, this is a conservative assumption, because the Salem CREACS HEPA filter is tested to a particulate removal efficiency of 99 percent. The licensee assumes control room occupancy factors of 100 percent for the first 24 hours, 60 percent from 24 to 96 hours, and 40 percent from 96 to 720 hours, which is consistent with the guidance of RG 1.183.

In support of the proposed TS change, the licensee also calculated the control room radiological dose consequences of an FHA occurring in containment and the FHB, utilizing the same accident scenarios and assumptions discussed above. The control room dose consequences of the postulated design-basis FHA, as described in Sections 8.1 through 8.3 of Attachment 5 to the licensee's application dated October 17, 2007, are shown in Table 4 of this SE.

The control room doses calculated by the licensee, resulting from the postulated design-basis FHA are below the regulatory dose criterion of 5.0 rem TEDE shown in 10 CFR 50.67 and GDC-19. The FHA in the FHB with rapid release proved to be the limiting event for all analyzed FHAs and decay times, resulting in a control room dose of 2.06 rem TEDE, assuming a 24-hour decay period.

3.1.2 FHA Radiological Consequences Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impact of implementing the proposed changes to the Salem Unit 2 TSs. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria identified in SE Section 2.0. In all cases, the calculated doses were below the applicable acceptance criteria. Based on the above findings, the NRC staff concludes that there is reasonable assurance that, in the event of a design-basis FHA following implementation of the proposed amendment, the EAB, LPZ, and control room doses would be below the applicable acceptance criteria. Therefore, the NRC staff concludes that the proposed amendment is acceptable with respect to radiological consequences.

3.2 SFP Cooling

Salem has two separate and independent SFPs, with one pool associated with each operating reactor. The proposed amendment involves a change in the minimum decay time for fuel removed from the Unit 2 reactor during outage 2R16, which increases the maximum decay heat generation that could be placed within the Unit 2 SFP during the first several hours of that refueling outage. Under otherwise identical conditions, a larger decay heat generation rate leads to a higher peak SFP temperature with the cooling systems in operation and a greater rate of temperature increase if the cooling system fails.

Decay heat is removed from each SFP by the spent fuel cooling system. Each spent fuel cooling system consists of one seismically-qualified cooling train, which includes two full-capacity pumps and one heat exchanger. The pumps are supplied power from the vital electrical distribution system that is backed by the emergency diesel generator. In the event one cooling system fails, the remaining system can be periodically swapped between the two SFPs to provide intermittent cooling to both pools. Heat is removed from the spent fuel cooling system heat exchangers by the safety-related component cooling water (CCW) system, which is cooled by the safety-related service water (SW) system.

The NRC staff reviewed the information provided by the licensee in support of the proposed license amendment, related to SFP cooling, as described below.

3.2.1 SFP Cooling Analysis

The proposed change to TS 3.9.3, which reduces the minimum decay time from 100 hours to 86 hours during refueling outage 2R16, will result in the potential for increased decay heat loads within the SFP. In its license amendment request dated October 17, 2007, PSEG described an evaluation of SFP cooling capability and procedural controls implemented through Salem's Integrated Decay Heat Management (IDHM) Program to assure spent fuel temperature limits are maintained. This program requires a pre-outage assessment of the SFP pool heat loads and heatup rates to assure availability of adequate SFP cooling capability prior to offloading fuel. As described in UFSAR Section 9.1.3, the procedural controls included in the IDHM Program require that, prior to each refueling, the following steps be taken:

(1) Calculate that the SFP water temperature will not exceed 149 °F following full core offload, using one heat exchanger for each SFP, and provide the required CCW temperature to achieve such results to the Salem Operations staff.

(2) Calculate that the SFP water temperature will not exceed 180 °F following full core offload, using one heat exchanger for both SFPs, and provide the required CCW temperature to achieve such results to the Salem Operations staff.

(3) Prior to initiating core offload activities, validate the assumptions used in the IDHM Program calculations. The validation includes ensuring the availability of both SFP heat exchangers and verifying that the actual CCW temperatures are consistent with the IDHM calculation input values.

In Attachment 4 to the licensee's application dated October 17, 2007, PSEG provided Calculation S-C-SF-MDC-1800, Revision 6, "Decay Heat-up Rates and Curves, Refueling Outage 2R16." This calculation provided the basis for the evaluation of SFP cooling capability for outage 2R16.

Calculation S-C-SF-MDC-1800 evaluated the SFP peak temperature by analysis using the licensee's computer program, CROSSTIE. In the NRC safety evaluation for Amendments 251 and 232 for Salem Units 1 and 2, respectively, the NRC staff accepted the use of CROSSTIE in the IDHM Program to provide an accurate representation of peak SFP temperature. The licensee's analyses used the following assumptions:

(1) A full core offload begins at the proposed minimum decay time of 86 hours and is completed 41 hours later.

(2) Decay heat (background heat in SFP plus offloaded core) is calculated based on the methodology given in NRC Branch Technical Position ASB 9-2. The calculated decay heat is based on actual fuel assembly burnups and is conservatively based on 100 percent capacity factor.

(3) Credit is taken for heat loss to the FHB atmosphere. The FHB ambient temperature is conservatively assumed to be 110 °F. The design value is 105 °F, which bounds the maximum calculated temperature based on the design SFP temperature of 180 °F. The FHB relative humidity is conservatively assumed to be 100 percent.

(4) The net SFP water volume is based on the minimum allowed SFP level of 23 feet above the fuel assemblies (in accordance with TS 3.9.11) and a fully offloaded core.

Using the above methods and assumptions, the licensee evaluated the following conditions:

(1) Normal cooling where each unit's SFP is aligned to its spent fuel cooling heat exchanger (SFCHX). This case determines the peak SFP temperature as a function of CCW temperature and the CCW temperature that results in a peak SFP temperature of 149 °F. The licensee determined a corresponding maximum SW temperature and compared it to the historical maximum for the associated time of year. For this case, the licensee assumed that the outage SFP is aligned to a single SFCHX. It also assumes that only one CCW heat exchanger is aligned, which lowers the maximum allowable SW temperature.

(2) Cross-connect operation where the SFPs for each unit are swapped between a single SFCHX, with the other unit's SFCHX unavailable (in the unlikely event that a SFCHX becomes unavailable post core off-load). This case determines the time available to swap cooling between the SFPs prior to the uncooled SFP reaching the design limit of 180 °F. For this case, the licensee conservatively assumed that a loss of cooling to one SFP occurs right after core offload is complete, with maximum decay heat load in the outage (Unit 2) SFP.

For the normal cooling conditions, the licensee determined that a CCW temperature of 88 °F results in a peak Unit 2 SFP temperature of 148.5 °F, which is just below the licensing basis limit of 149 °F. Under typical refueling conditions, the licensee stated that this CCW temperature correlates to a maximum SW temperature of 79 °F. The licensee determined that the historical maximum SW temperature for March is approximately 53 °F. Consequently, the SW temperature for 2R16 will likely be much less than 79 °F and, at credible SW temperatures during 2R16, substantial margin above the cooling capability necessary to maintain the SFP temperature below the limit of 149 °F will be available.

For cross-connect operations, the licensee determined that both SFPs can be maintained below the licensing basis limit of 180 °F with a single available SFCHX. For loss of the non-outage (Unit 1) SFCHX, the licensee determined the Unit 1 SFP would reach a peak temperature of 178 °F at about 120 hours after the Unit 2 core is offloaded with no forced cooling. Therefore, the licensee's analysis indicates that restoration of forced cooling to the Unit 1 SFP would not be necessary to maintain the peak temperature of the Unit 1 SFP below the licensing basis limit. For loss of forced cooling to the outage (Unit 2) SFP, the licensee determined that the time to reach the licensing basis SFP temperature limit of 180 °F would be 3.4 hours from SFP conditions expected at the end of the offload with a CCW temperature of 70 °F. The time would decrease to 2.7 hours for a CCW temperature of 80 °F, and decrease to 1.3 hours for a CCW temperature of 99 °F. The licensee noted that CCW temperatures above 88 °F would be precluded by procedural controls applied to refueling operations through the IDHM Program based on the evaluation of normal refueling conditions. The licensee validated through walkdowns that realignment of a spent fuel cooling system loop to the alternate pool requires less than one hour. The licensee stated that procedures direct resetting of the SFP high temperature alarm for SFP temperatures above the normal setpoint of 125 °F to maintain effective indication of a loss of cooling. Therefore, in the unlikely event of a loss of cooling to the Unit 2 SFP near the completion of the offload, adequate time would be available to identify the loss of cooling and realign cooling from the Unit 1 SFP to the Unit 2 SFP before the Unit 2 SFP temperature exceeds the licensing basis temperature limit of 180 °F. The licensee also evaluated the potential repeated swapping of spent fuel cooling between pools to maintain both SFPs below 170 °F and determined that adequate time to realign cooling would be available for this more restrictive control of peak SFP temperature.

The licensee presented validation data from previous refueling outages at SFP temperatures ranging from 66 °F to approximately 120 °F. This information confirmed the accuracy of the CROSSTIE program at temperatures where evaporative cooling from the SFP is unimportant. However, the staff questioned the accuracy of the program in modeling heat transfer to the FHB when evaporative cooling is important, such as scenarios involving loss of forced cooling. To address this issue, the license presented sensitivity analyses demonstrating that the SFP heat-up projections from the CROSSTIE model were relatively insensitive to FHB temperatures between 105 °F and 120 °F.

After reviewing these projections, the NRC staff requested that the licensee address how the CROSSTIE model provides a realistic estimate of heat loss to the FHB. In the supplement dated January 11, 2008, the licensee explained that the dominant mode of heat transfer is through convective heat loss and that the rate of heat transfer is dependent on the difference between the vapor pressure of water at the pool surface temperature and the vapor pressure in the ambient FHB atmosphere. For pool temperatures above 150 °F, the licensee has determined that the FHB ventilation system must be operating to maintain the FHB building atmosphere below design conditions of 100 percent humidity at 105 °F. These conditions maintain the necessary difference in vapor pressure between the pool surface and the building atmosphere to provide the assumed evaporative cooling. The licensee evaluated the effect of elevated pool temperatures on the FHB building ventilation system and concluded that the FHB ventilation system could accommodate expected condensation at SFP temperatures up to 180 °F. Therefore, although the CROSSTIE model of evaporative cooling is not validated at high SFP temperatures, the methodology and assumptions are adequate to provide reasonable estimates of the time for pool temperature to increase to specific values. The available margin in time to reestablish cooling following loss of forced cooling and the additional margin provided by the expected low SW temperatures provide reasonable assurance that the licensing basis SFP temperature limits will be maintained in the unlikely event of a loss of forced cooling to either SFP.

3.2.2 SFP Cooling Conclusion

Based on the review as described in SE Section 3.2.1, the NRC staff concludes that the licensee's analysis associated with reducing the minimum decay time from 100 hours to 86 hours on a one-time basis for refueling outage 2R16, in conjunction with the IDHM Program procedural controls as described in UFSAR Section 9.1.3, provide reasonable assurance that the available decay heat removal capability will be maintained consistent with its importance to safety and that the SFP cooling system will provide the capability to prevent a significant reduction in coolant inventory under accident conditions. Specifically, the decay heat removal capability is acceptable because: (1) the SFP cooling system will be capable of maintaining an appropriate pool temperature consistent with the current licensing basis during planned refueling evolutions; and (2) with the failure of a single cooling train, the cooling system will maintain SFP temperature within analyzed limits for SFP structural integrity with the remaining cooling system in operation to cool both trains. Therefore, the proposed amendment is acceptable with respect to SFP cooling.

The NRC staff also concludes that the provisions of 10 CFR 50.59 provide adequate control for any future changes to the IDHM Program procedural controls currently described in UFSAR Section 9.1.3. Specifically, 10 CFR 50.59(c)(2)(viii) requires that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90, prior to implementing a proposed change, if the change would "[r]esult in a departure from a method of evaluation described in the Final Safety Analysis Report (as updated) used in establishing the design basis or in the safety analyses." The definition in 10 CFR 50.59(a)(2) states that departure from a method of evaluation described in the Final Safety Analysis Report (as updated) used in establishing the design basis or in the safety analyses means: (i) changing any of the elements of the method described in the FSAR unless the results of the analysis are conservative or essentially the same[2]; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by the NRC for the intended application. The IDHM procedural controls currently described in UFSAR Section 9.1.3 (and discussed above in SE Section 3.2.1) provide a method of evaluation acceptable to the NRC staff to assure the availability of adequate SFP cooling capability prior to offloading fuel. Change to this method of evaluation is subject to the provisions of 10 CFR 50.59(c)(2)(viii).

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2 As discussed in Section 3.4 of Nuclear Energy Institute NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, gaining margin by revising an element of a method of evaluation is considered to be a non-conservative change. NEI 96-07, Revision 1 was endorsed by NRC Regulatory Guide 1.187 as providing methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59.

3.3 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 and 3.2, 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 amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 68218). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Attachment: Tables 1 - 9

TABLE 1
Licensee Calculated Off-Site Dose Results for Fuel Handling Accidents In Containment

Fuel Decay Time (hr)	EAB (rem TEDE)	LPZ (rem TEDE)	Dose Criteria (rem TEDE)
24	1.26	0.18	6.3
48	1.05	0.15	6.3
60	0.99	0.14	6.3
72	0.93	0.13	6.3
96	0.84	0.12	6.3

TABLE 2
Licensee Calculated Off-Site Dose Results for Fuel Handling Accidents In Fuel Handling Building With Failure of an Exhaust Fan (2-Hour Release)

Fuel Decay Time (hr)	EAB (rem TEDE)	LPZ (rem TEDE)	Dose Criteria (rem TEDE)
24	1.26	0.18	6.3
48	1.05	0.15	6.3
60	0.99	0.14	6.3
72	0.93	0.13	6.3
96	0.84	0.12	6.3

TABLE 3
Licensee Calculated Off-Site Dose Results for Fuel Handling Accidents In Fuel Handling Building With Rapid Release of 1 Volume / Min

Fuel Decay Time (hr)	EAB (rem TEDE)	LPZ (rem TEDE)	Dose Criteria (rem TEDE)
24	1.27	0.18	6.3
48	1.06	0.15	6.3
60	1.00	0.14	6.3
72	0.94	0.13	6.3
96	0.85	0.12	6.3

TABLE 4
Licensee Calculated Control Room Dose Results for Fuel Handling Accidents

Fuel Decay Time (hr)	FHA in Containment (rem TEDE)	FHA in FHB With Fan Failure (rem TEDE)	FHA in FHB With Rapid Release (rem TEDE)	Dose Criteria (rem TEDE)
24	1.13	0.73	2.06	5.0
48	0.95	0.62	1.78	5.0
60	0.89	0.58	1.67	5.0
72	0.84	0.55	1.58	5.0
96	0.76	0.49	1.43	5.0

TABLE 5
Licensee FHA Analysis Assumptions

Reactor power	3632 MWt
Radial peaking factor	1.70
Fission product decay period, FHA in containment or in FHB (SFP)	24, 48, 60, 72, and 96 hours
Number of fuel assemblies in core	193
Number of fuel assemblies damaged	1
Number of fuel rods damaged per assembly, FHA in containment or in SFP	All
Fuel Gap fission product inventory: I-131 Kr-85 Other halogens and noble gases Alkali metals	 16% 20% 10% 24%
Iodine species fractions in pool water: Elemental Organic Particulates	 0.9985 0.0015 none
Water depth above damaged fuel	23 ft
Pool iodine effective decontamination factor	200
Chemical form of iodine above SFP: Elemental Organic	 57% 43%
Release modeling: Containment Release – 2-hour release period FHB Release – 2-hour release period FHB Release – “Rapid” Release	
Control room envelope volume	81,420 ft^3
Control room normal flow rate	1,320 cfm
CREACS ventilation flow rate	8000 cfm

TABLE 5 (continued)
Licensee FHA Analysis Assumptions

CREACS System start delay time, minutes	1 minute
Control room unfiltered inleakage with the CREACS in operation	150 cfm
CREACS filter efficiency: Charcoal Filter HEPA Filter	 95% 95%
Control room occupancy factors: 0 - 24 hr 24 - 96 hr 96 - 720 hr	 1.0 0.6 0.4
Control room breathing rate	3.5 E-4 m^3/sec
EAB Atmospheric Dispersion Factor (X/Q)	1.30E-4 sec/m^3
LPZ Atmospheric Dispersion Factors (X/Qs): 0 - 2 hr 2 - 8 hr 8 - 24 hr 24 - 96 hr 96 - 720 hr	 1.86E-5 sec/m^3 7.76E-6 sec/m^3 5.01E-6 sec/m^3 1.94E-6 sec/m^3 4.96E-7 sec/m^3
Offsite Breathing Rate: 0 - 8 hr 8 - 24 hr 24 - 720 hr	 3.5E-4 m^3/sec 1.8E-4 m^3/sec 2.3E-4 m^3/sec

TABLE 6

Control Room Atmospheric Dispersion Factors

Post-FHA Release from Containment Equipment Hatch

(χ/Q values in sec/m^3)

Time Interval	χ/Q Value
0 - 2 hrs	2.86E-3
2 - 8 hrs	2.22E-3
8 - 24 hrs	9.15E-4
24 - 96 hrs	6.60E-4
96 - 720 hrs	5.62E-4

TABLE 7

Control Room Atmospheric Dispersion Factors

Post-FHA Release from Plant Vent

(χ/Q values in sec/m^3)

Time Interval	χ/Q Value
0 - 2 hrs	1.78E-3
2 - 8 hrs	1.31E-3
8 - 24 hrs	5.22E-4
24 - 96 hrs	3.77E-4
96 - 720 hrs	3.17E-4

TABLE 8
Control Room Atmospheric Dispersion Factors
Post-FHA Release from FHB Roll-up Door
(χ/Q values in sec/m^3)

Time Interval	χ/Q Value
0 - 2 hrs	1.50E-3
2 - 8 hrs	1.20E-3
8 - 24 hrs	4.48E-4
24 - 96 hrs	3.22E-4
96 - 720 hrs	2.50E-4

TABLE 9
Control Room Atmospheric Dispersion Factors
Post-FHA Release from Smoke Hatch
(χ/Q values in sec/m^3)

Time Interval	χ/Q Value
0 - 2 hrs	1.15E-2
2 - 8 hrs	9.28E-3
8 - 24 hrs	3.50E-3
24 - 96 hrs	2.49E-3
96 - 720 hrs	2.02E-3