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LR-N06-0309
LCR S06-07

PSEG
Nuclear LLC

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

**REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
REFUELING OPERATIONS -DECAY TIME
SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311**

References: (1) Letter from PSEG to NRC: "Request for Changes to Technical Specifications, Refueling Operations – Fuel Decay Time Prior to Core Alterations or Movement of Irradiated Fuel, Salem Nuclear Generating Station, Units 1 and 2, Facility Operating Licenses DPR-760 and DPR-75, Docket Nos. 50272 and 50-311", dated June 28, 2002

(2) Letter from PSEG to NRC: "Additional Information – Spent Fuel Pool Cooling, Request for License Amendment, Refueling Operations – Fuel Decay Time Prior to Commencing Core Alterations or Movement of Irradiated Fuel, Salem Nuclear Generating Station, Units 1 and 2, Facility Operating Licenses DPR-760 and DPR-75, Docket Nos. 50272 and 50-311", dated October 2, 2002

(3) Letter NRC to PSEG, "Salem Nuclear Generating Station, Unit Nos 1 and 2, Issuance of Amendment Re: Refueling Operations – Fuel Decay Time Prior to Commencing Core Alterations or Movement of Irradiated Fuel (TAC NOS. MB5488 AND MB5489)" dated October 10, 2002

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment of the Technical Specifications (TS) for the Salem Nuclear Generating Station, Units 1 and 2. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG proposes to revise the requirements for Fuel Decay Time prior to commencing movement of irradiated fuel. TS 3/4.9.3 "Decay Time" is revised to allow fuel movement in the containment to commence at the time calculated using the Salem Spent Fuel Pool Integrated Decay Heat Management (IDHM) Program. This program, which is part of the Salem Outage Risk Management Program, will provide a conservative fuel movement time and assure available spent fuel pool cooling capability prior to fuel

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movement. TS 3/4.9.3 will also be revised to include a limitation on Fuel Decay Time based on the Fuel Handling Accident (FHA) analyses

Currently, TS 3/4.9.3 is configured on a broad calendar approach; fuel decay time is longer between May 16th and October 14th (168 hours decay time limit) versus the time between October 15th and May 15th (100 hours) based on gross average river water temperature which is significantly cooler in the fall and winter months. This calendar approach was previously approved for Salem Units 1 and 2 via Amendments 251 and 232 (Reference 3), based on information provided in References 1 and 2. This new request follows the same basic approach as evaluated in References 1 and 2, but is based on additional analyses and refinements. The proposed TS change to use the IDHM Program will result in an outage specific fuel decay time that provides a higher level of precision and accuracy, and ensures the necessary parameters are conservatively evaluated.

This proposed change, which is discussed in detail in Attachment 1, is based on the following:

1. A re-analysis of the FHA radiological consequences applying the guidelines contained in 10CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term.

2. A Spent Fuel Pool (SFP) re-analysis to determine the capability of the cooling system to maintain Spent Fuel Pool water temperature below the analyzed limit of 180°F (with one heat exchanger in operation) and 149°F with two heat exchanger operation.

3. A PSEG commitment to use the Salem Spent Fuel Pool Integrated Decay Heat Management (IDHM) Program[1] calculation methodology prior to each Salem refueling to:

    i. Calculate that the SFP temperature will not exceed 149°F following full core offload, using one heat exchanger per SFP (two heat exchangers total) and to provide to the Operations staff the required Component Cooling Water temperature to achieve such results.
    ii. Calculate that the SFP temperature will not exceed 180°F following full core offload with one heat exchanger available for both SFP's and to provide to the Operations staff the required Component Cooling Water temperature to achieve such results.

Attachment 2 provides the existing TS pages marked-up to show the proposed changes Attachment 3 summarizes the regulatory commitments made in this submittal. For your

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[1] The Spent Fuel Pool Integrated Decay Heat Management Program, described in Salem LCR S02-03 (References 1 and 2) is designed to perform heat load calculations prior to core fuel offloads to establish the fuel decay time and ensure the required cooling is available such that the Spent Fuel Pool water temperature limits are not exceeded.

information, Attachment 4 provides the existing TS Bases pages marked-up to reflect the associated changes to the TS. Attachment 5 provides the SFP Cooling System Capability Evaluation (Calculation S-C-SF-MEE-1679, Rev. 1). Attachment 6 provides the Fuel Handling Accident (FHA) Radiological Consequences Evaluation (Calculation S-C-ZZ-MDC-1920, Rev. 4IR0)

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. This amendment to the Salem TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

PSEG requests approval of the proposed License Amendment by March 1, 2007 to be implemented within 60 days, to support Salem Unit 1 refueling outage 1R18.

Should you have any questions regarding this request, please contact Mr. Jamie Mallon at 610-765-5507.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Executed on 8/4/06

Thomas P. Joyce
Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachments (6)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
REFUELING OPERATIONS – FUEL DECAY TIME

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS REFUELING OPERATIONS – FUEL DECAY TIME

1. DESCRIPTION

Currently TS 3.9.3 requires that: (a) the reactor has been subcritical for at least 100 hours prior to movement of irradiated fuel in the reactor vessel between October 15th through May 15th, and (b) the reactor has been subcritical for at least 168 hours prior to movement of irradiated fuel in the reactor vessel between May 16th and October 14th. The action statement requires suspension of all operations involving movement of irradiated fuel within the reactor pressure vessel with a decay time of less than the specified calendar requirements. The associated surveillance, Surveillance Requirement (SR) 4.9.3 requires verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

TS 3.9.3 would be changed by eliminating the current (a) and (b) broad calendar provision (which is based on peak river water temperature) and replacing it with a more precise determination of spent fuel pool (SFP) cooling capability, using the Salem Spent Fuel Pool Integrated Decay Heat Management (IDHM) Program[2] calculation methodology prior to each Salem refueling. SR 4.9.3 will remain unchanged.

2. PROPOSED CHANGE

TS 3.9.3 would be revised as follows:

"The reactor shall be subcritical for at least:

The minimum decay time for the movement of fuel as determined by the SFP Integrated Decay Heat Management (IDHM) Program*, and shall not be less than 24 hours**.

> *The IDHM program will establish the minimum in-vessel decay time needed to assure the SFP limits of 149°F with two available heat exchangers and 180°F with only one heat exchanger prior to the start of each specific Salem refueling outage.
>
> **The current radiological design bases analysis for the Fuel Handling Accident (FHA) is based on a minimum decay time of 24 hours prior to movement of irradiated fuel assemblies within the reactor vessel. Therefore, the decay time for movement of fuel cannot be less than 24 hours."

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2 The Spent Fuel Pool Integrated Decay Heat Management Program, described in Salem LCR S02-03 (PSEG letters LR-N02-0231 dated June 28, 2002 and LR-N02331 dated October 10, 2002) is designed to perform heat load calculations prior to core fuel offloads to establish the fuel decay time and ensure the required cooling is available such that the Spent Fuel Pool water temperature limits are not exceeded.

3. BACKGROUND

The 100-hour decay time requirement between October 15th through May 15th was included in the Salem TS via Amendments 251 and 232 to DPR-70 and DPR-75, respectively, on October 10, 2002. This change was requested because the 168-hour requirement conservatively covered the entire year; it imposed an unnecessary penalty on plant operators in the cooler months, when refuelings are typically scheduled. The NRC staff concluded that this change was acceptable based (1) on the analysis provided for the SFP cooling capability, (2) the radiological consequences of a FHA, and (3) the use of the IDHM Program.

For SFP cooling capability, the NRC staff concluded that the proposed revisions to TS 3/4.9.3, in conjunction with the specified operational controls, ensure that the available decay heat removal capability will be maintained consistent with its importance to safety and that the SFP cooling system provides 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 design 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.

For the FHA analysis, the NRC staff found that PSEG used analysis methods and assumptions consistent with the conservative guidance of RG 1.183. The staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff concluded that the estimates of the TEDE due to FHA accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

For the IDHM Program, the staff reviewed the critical software document (Crosstie) and audited the decay heat management program evaluation of the most recent Salem refueling outage. The staff found that the software was calibrated against actual plant data, and that validation against SFP transients, including loss and restoration of cooling, produced acceptable agreement. The staff also found that, with representative input data, the predicted SFP temperature profile was in close agreement with that of the previous refueling outage. The SER for Amendments 251 and 232 states: "Accordingly, the staff concluded that the Crosstie software, as implemented in the decay heat management program, will provide an accurate representation of peak SFP temperature."

In-vessel decay is required before moving a fresh, hot core into the SFP because of the radiation dose and fuel-pool cooling requirements. With regard to pool

cooling, decay heat from previously irradiated fuel elements constantly decreases as the fission products and heavy elements decay. Therefore, the longer the elements are allowed to decay within the reactor vessel, the less heat duty is transferred to the SFP.

The Salem UFSAR, Section 9.1.3.1 makes the following statements:

"The Spent Fuel Pool Cooling System maintains pool temperature at or below 149°F, provided both SFP heat exchangers are available. If only one heat exchanger is available, pool temperature is limited to 180°F."

Later, in Section 9.1.3.2, the UFSAR states:

"In 1998, additional spent fuel pool heat removal analyses were performed. The analyses addressed potential full-core off-loads during upcoming refueling outages as well as end of plant life. These analyses concluded one pump and one heat exchanger can maintain pool temperature below 149°F under all combinations of decay time and Component Cooling Water (CCW) temperature except minimum decay times and very high cooling water temperatures. Under these later conditions, in vessel decay-time would be extended or parallel heat exchanger operation would be used to maintain pool temperature below 149°F."

In addition to the above, Section 9.1.3.2 describes the SFP IDHM program under which pre-outage assessments of SFP heat loads are performed prior to core offload as follows:

- Calculations to assure SFP temperature does not exceed 149°F following a full-core offload with one heat exchanger per pool.
- Calculations to assure SFP temperature does not exceed 180°F following a full-core offload with one heat exchanger for both pools.
- Validation of assumptions in the Integrated Decay Heat Management program including
    - Availability of both heat exchangers, each with an available pump and
    - Actual CCW system temperatures consistent with calculated values.

4. TECHNICAL ANALYSIS

In order to evaluate the acceptability of using the IDHM Program to calculate the fuel decay time prior to commencing movement of irradiated fuel, two evaluations were performed: Spent Fuel Pool Cooling Capacity and Fuel Handling Accident (FHA) Radiological Consequences. The purpose of these two evaluations was to determine which was the more limiting for fuel movement. Summaries of these two evaluations are provided below:

4.1 Spent Fuel Pool Cooling System Capability

The SFP cooling minimum time restriction will occur during the cooler months of the year. An evaluation (Attachment 5) was performed, applicable to both Salem Unit 1 and Salem Unit 2, that addressed the period from October 15^{th} through May 15^{th}, which is typically when refueling outages are scheduled (i.e., in the spring and fall). During this period CCW temperature is expected to be 71°F or below.

The evaluation considers heat removal from the Salem Spent Fuel Pools using forced cooling provided by the SFPC heat exchangers. By relying only on the SFPC heat exchangers, the analysis contains several substantial conservatisms as described below. These conservatisms could be credited in this calculation. However, at this time they will be left as providing additional temperature margins.

1 No credit is taken for evaporative cooling, i.e. pool bulk temperature cooling resulting from evaporation at the surface of the SFP, provided that both SFP heat exchangers are available[3]. Reference 5.5 of Attachment 5 indicates that evaporative cooling contributes 0.86×10^6 Btu/hour at 150 °F and 3.87×10^6 Btu/hour at 180 °F. In addition, another 311,800 Btu/hr are consumed in heating the cold makeup water that replaces the evaporation (from 100°F to 180°F). Consequently, if the pool reaches 180°F, evaporative cooling plus makeup heating removes approximately 9% of the peak heat in the hot pool and approximately 49% of the heat in the non-refueling pool.

2 No credit is taken for cooling through the concrete structure of the pool. Heat is conducted through the pool steel liner, concrete structure, and ultimately to the cooler environment beyond the structure. The higher the pool water temperature, the more heat transmitted through the structure.

3 RHR cooling continues to provide cooling to the SFP with all fuel elements removed to the SFP as long as the refueling canal remains flooded and the transfer gate is open. The cooler water in the reactor vessel and refueling canal will transfer to the SFP via natural circulation through the transfer gate. This potential cooling source is never credited in any analysis or procedure.

These inherent conservatisms are of sufficient magnitude to account for any foreseeable changes in river temperatures or other potentially non-conservative assumptions. In addition, a more accurate assessment of the pool heat up times is required to be performed prior to a refueling outage, as part of the Integrated Decay Heat Management Program.

The evaluation (Attachment 5) demonstrates that a fully radiated 193 element reactor core can be off-loaded to either Salem spent fuel pool with 85-hours of in-vessel decay provided the CCW outlet temperature is less than or equal 71°F.

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[3] In the abnormal case where only one heat exchanger is available for both fuel pools, evaporative cooling from the pool surface will be considered in order to determine more realistic timing for the heat exchanger transfer between pools.

The evaluation also demonstrates that the required temperature (less than or equal 71°F) can be expected during the period October 15th through May 15th, annually.

Consequently, a decay time of 85 hours is what will be typically required for refueling outages in the spring and fall. If CCW temperature is slightly above, or slightly below 71°F during the scheduled outage, then the decay time will be slightly more or slightly less that 85 hours.

This conclusion is based on the capability of the SFP cooling system to (1) maintain both Salem pools below 149°F with two SFPC heat exchangers available and (2) maintain both pools below 180°F with only one heat exchanger available. This capability meets the requirements of UFSAR Chapter 9.1.3.1.

4.2 Fuel Handling Accident Radiological Consequences

The purpose of this analysis is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and Control Room (CR) doses due to a fuel handling accident (FHA) occurring in the containment building and in the Fuel Handling Building (FHB). The FHA analyses are performed using the Alternative Source Term (AST), guidance in the Regulatory Guide 1.183, Appendix B, and TEDE dose criteria. Additional conservatism was used by assuming no containment closure during fuel movement; activity is released to the environment through the opened Containment Equipment Hatch (CEH) or the plant vent (PV).

The regulatory requirements in the Regulatory Guide 1.183, Appendix B are adopted as assumptions, which are incorporated as design inputs along with other plant-specific as-built design parameters.

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

The results of the evaluation (Attachment 6) demonstrate that the irradiated fuel can be moved after the reactor vessel has been sub-critical for 24-hours. The doses shown in Attachment 6 of this application are less than the TEDE criteria set forth in RG 1.183 and are a small fraction of the dose criteria in 10CFR 50.67.

4.3 Use of the Integrated Decay Heat Management Program

The results of Sections 4.1 and 4.2 demonstrate that the SFP cooling capability is the more limiting parameter, versus FHA evaluation, for the decay time for fuel movement. Consequently, the TS for decay time will utilize the SFP IDHM Program, with a lower limit restriction based on the FHA evaluation.

The Integrated Decay Heat Management (IDHM) program calculates peak SFP temperature for each refueling outage using the Crosstie computer program.

To support LCR S02-03 that reduced the "time to move fuel after shutdown" from 168 hrs to 100 hrs, the calculated 2R12 SFP heatup data (Crosstie) was benchmarked to actual SFP temperature data obtained during 2R12. The results of the IDHM program calculations showed substantial conservatism when predicted SFP temperatures were compared to actual SFP temperatures recorded during the outage. Actual recorded values of Component Cooling Water (CCW) supply temperatures were used as input to the IDHM calculation for 2R12, resulting in calculated SFP temperatures that closely correlated with actual SFP temperatures; and provided validation of the ability of the IDHM calculations to accurately predict maximum SFP temperatures. The decay heat loads used for the 2R12 IDHM program predictions are indicative of a typical 18-month production run, and subsequent refueling outage decay heat loads would not vary significantly. Consequently, the IDHM will provide a precise and accurate SFP decay heat-up, while maintaining the inherent conservatisms of the SFP Cooling capability. The NRC Staff concurred with this conclusion in the SER for Salem Amendments 251 and 232.

As part of implementation of the requested amendment, PSEG commits to using the IDHM program calculation methodology prior to each Salem refueling to:

- Calculate that the SFP temperature will not exceed 149°F following full core offload, using one heat exchanger per SFP (two heat exchangers total) and to provide to the Operations staff the required Component Cooling Water temperature to achieve such results.

- Calculate that the SFP temperature will not exceed 180°F following full core offload with one heat exchanger available for both SFP's and to provide to the Operations staff the required Component Cooling Water temperature to achieve such results.

PSEG Procedure SC.OM-AP.ZZ.0001, Shutdown Safety Management Program – Salem Annex, implements the IDHM process for each refueling outage. PSEG Procedure S1/2.OP-IO.ZZ-0007 then validates the IDHM results, verifying actual CCW temperature and decay time.

PSEG procedure S1(2).OP-IO.ZZ-0010(Q), Spent Fuel Pool Manipulations, establishes SFP cooling requirements. The availability of the Spent fuel cooling system is then monitored using the Outage Risk Assessment Model (ORAM) logic and Outage Risk Assessment procedure OU-AA-103(Q).

The Salem Unit 1 and 2 SFP high temperature alarm setpoint is 125°F. The alarm setpoint is also an entry condition for abnormal operating procedure S1(2).OP-AB.SF-0001(Q), Loss of Spent Fuel Pool Cooling. If peak SFP temperature, as predicted by the IDHM program, exceeds 125°F for a refueling outage, then exceeding the alarm setpoint is an expected condition, and the alarm would not be indicative of an actual loss or degradation of SFP cooling. However, the SFP high temperature alarm capability will alert the operators in the

event that SFP temperature exceeds the peak temperature predicted by the IDHM program for each refueling outage.

5.0 Regulatory Safety Analysis

5.1 Basis for proposed no significant hazards consideration determination

As required by 10 CFR 50.91(a), PSEG provides its analysis of the no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
2. create the possibility of a new or different kind of accident from any previously analyzed; or
3. involve a significant reduction in a margin of safety.

The determinations that the criteria set forth in 10 CFR 50.92 are met for this amendment request are indicated below:

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   Response: No.

   The proposed license amendment would allow fuel assemblies to be removed from the reactor core and be stored in the Spent Fuel Pool in less time after subcriticality (but more accurately calculated), than currently allowed by the TSs. Decreasing the decay time of the fuel affects the radionuclide make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated. The accident previously evaluated that is associated with the proposed license amendment is the fuel handling accident. Allowing the fuel to be offloaded based on the IDHM calculated time after subcriticality does not impact the manner in which the fuel is offloaded. The accident initiator is the dropping of the fuel assembly. Since earlier offload does not affect fuel handling, there is no increase in the probability of occurrence of a fuel handling accident. The time frame in which the fuel assemblies are moved has been evaluated against the 10 CFR 50.67 dose limits for members of the public, licensee personnel and control room. Additionally, the guidance provided in Reg. Guide 1.183 was used for the selective application of Alternative Source Term. All dose limits are met with the reduced core offload times; and significant margin is maintained, as the minimum decay time prior to movement of fuel for the FHA analysis is 24 hours.

Therefore, the proposed license amendment does not increase the probability of occurrence or the consequences of accidents previously evaluated are not increased.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   Response: No.

   The proposed license amendment would allow core offload to occur in less time after subcriticality (but more accurately calculated), which affects the radionuclide make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The radionuclide makeup of the fuel assemblies and the amount of decay heat produced by the fuel assemblies do not currently initiate any accident. A change in the radionuclide makeup of the fuel at the time of core offload or an increase in the decay heat produced by the fuel being offloaded will not cause the initiation of any accident. The accident previously evaluated that is associated with fuel movement is the fuel handling accident. There is no change to the manner in which fuel is being handled or in the equipment used to offload or store the fuel. The effects of the additional decay heat load have been analyzed. The analysis demonstrated that the existing Spent Fuel Pool cooling system and associated systems under worst-case circumstances would maintain the integrity of the Spent Fuel Pool. The proposed method of offload does not create a new or different kind of accident from any accident previously evaluated.

   Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   Response: No.

   The margin of safety pertinent to the proposed changes is the dose consequences resulting from a fuel handling accident. The shorter decay time prior to fuel movement has been evaluated against 10 CFR Part 50.67 and all limits continue to be met. All dose limits are met with the reduced core offload times; and significant margin is maintained, as the minimum decay time prior to movement of fuel for the FHA analysis is 24 hours. Decay heatup calculations performed prior to each refueling outage as part of the IDHM program ensure that planned spent fuel transfer to the SFP will not result in maximum SFP temperature exceeding the design basis limit of 149°F (with both heat exchangers available) or 180°F (with one heat exchanger alternating between the two pools). As stated above, the changes in radionuclide makeup and additional heat load do not impact any safety settings and do not cause any safety limit to not be met. In addition, the integrity of the Spent Fuel Pool is maintained.

The time frame in which the fuel assemblies are moved has been evaluated against the 10 CFR 50.67 dose limits for members of the public, licensee personnel and control room. Additionally, the guidance provided in Reg. Guide 1.183 was used. Calculations performed conclude that expected dose limits following a Fuel handling Accident are met with the proposed decay time prior to commencing fuel movement.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on this review, it is concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, PSEG proposes that a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

The NRC's traditional methods for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the ASTs and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in other regulatory documents when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

This application and the supporting analyses comply with this guidance as it applies to a Fuel Handling Accident.

Title 10, Code of Federal Regulations, Part 50 Section 67, "Accident Source Term".

10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design basis radiological consequences analyses. This document is part of a 10 CFR 50.90 license amendment application and evaluates the consequences of a design basis fuel handling accident as previously described in the Salem UFSAR.

USNRC Branch Technical Position ASB 9-2, Residual Decay Heat for Light-Water Reactors for Long-Term Cooling, Revision 2 of July 1981.

BTP ASB 9-2 uses a conservative approach for calculating fuel element decay heat, and is applied to this amendment without scaling factors or other adjustments.

Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors".

RG 1.183 supersedes corresponding radiological assumptions provided in other regulatory guides and standard review plan chapters when used in conjunction with an approved alternative source term and the TEDE provided in 10 CFR 50.67.

10 CFR 100, "Determination of Exclusion Area, Low Population Zone and Population Center Distance".

10 CFR 100.11 provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67. In accordance with the provisions of 10 CFR 50.67(a), PSEG applied the dose reference values in 10 CFR 50.67 (b) (2) in the analyses in lieu of 10 CFR 100 for the Fuel Handling Accident.

NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents".

The SRP Section 15.7.4 describes the radiological effects of a postulated Fuel Handling Accident. The SRP does not directly refer to the guidance of RG 1.183 or 10 CFR 50.67. Instead, it refers to regulatory documents, which are superseded by the selective application of the Alternative Source Term for the Fuel Handling Accident.

10 CFR 50 Appendix A, General Design Criteria 19, Control Room

PSEG has applied the guidelines provided by 10 CFR 50.67 and RG 1.183, which supersedes the current requirements of GDC 19 for the Fuel Handling Accident.

5.3 Conclusion

The FHA dose analyses were performed in accordance with AST and TEDE guidelines provided in Regulatory Guide 1.183 and 10 CFR 50.67. The assumptions and design inputs are listed in Engineering Calculations listed in the reference section. The SFP Cooling Capacity calculations were performed applying acceptable NRC guidance and conservatism aspects resulting in assurance that the design basis limits for SFP heat removal are maintained. Use of the IDHM Program will ensure that the decay time fuel movement limit is conservative.

The results of these analyses indicate that the doses shown in Attachment 6 of this application are less than the TEDE criteria set forth in RG 1.183 and are a small fraction of the dose criteria in 10CFR 50.67.

In conclusion, based on the considerations discussed above,

(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.

6.0 ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

PSEG has concluded that implementation of this amendment will have no adverse impact upon the Salem units; neither will it contribute to any significant additional quantity or type of effluent being available for adverse environmental impact or personnel exposure. The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite. The new consequences of the revised Fuel Handling Accident analysis remain well below the acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183.
It has been determined there is:

1. No significant hazards consideration,
2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and
3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Salem TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

7.0 REFERENCES

7.1 PSEG Calculation S-C-ZZ-MDC-1920, Revision 4IR0
7.2 PSEG Calculation S-C-SF-MEE-1679, Revision 1
7.3 PSEG Salem Units 1 and 2, Final Safety Analysis Report
7.4 PSEG Salem Units 1 and 2, Technical Specifications
7.5 PSEG, Shutdown Safety Management Program, OU-AA-103(Q)
7.6 PSEG Salem Units 1 and 2, S1(2).OP-IO.ZZ-0010(Q), Spent Fuel Pool Manipulations
7.7 PSEG Procedure, Salem Units 1 and 2, S1(2).OP-AB.SF-0001(Q), Loss of Spent Fuel Pool Cooling
7.8 PSEG Procedure, SC.OM-AP.ZZ.0001, Shutdown Safety Management Program – Salem Annex
7.9 PSEG Procedure, Salem Units 1 and 2, S1/2.OP-IO.ZZ-0007, Cold Shutdown to Refueling
7.10 10 CFR 50.67, "Accident Source Term"
7.11 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
7.12 NRC Branch Technical Position ASB 9-2 Revision 2 of July 1981, USNRC Standard Review Plan 9.2.5, Ultimate Heat Sink, NUREG 0800

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License DPR-70 are affected by this change request:

Technical Specification	Page
3/4.9.3	3/4 9-3

The following Technical Specifications for Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
3/4.9.3	3/4 9-3

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least:.

The minimum decay time for the movement of irradiated fuel in the reactor pressure vessel as determined by the SFP Integrated Decay Heat Management (IDHM) Program*, and shall not be less than 24 hours**.

*The IDHM program will establish the minimum in-vessel decay time needed to assure the SFP limits of 149°F with two available heat exchangers and 180°F with only one heat exchanger prior to the start of each specific Salem refueling outage.

**The current radiological design bases analysis for the Fuel Handling Accident (FHA) is based on a minimum decay time of 24 hours prior to movement of irradiated fuel assemblies within the reactor vessel. Therefore, the decay time for movement of fuel cannot be less than 24 hours.

~~a. 100 hours - Applicable through year 2010.~~
~~b. 168 hours~~

APPLICABILITY: Mode 6

~~Specification 3.9.3.a - From October 15th through May 15th, during movement of irradiated fuel in the reactor pressure vessel.~~

~~Specification 3.9.3.b - From May 16th through October 14th, during movement of irradiated fuel in the reactor pressure vessel.~~

ACTION:

With the reactor subcritical for less than the required time, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical as required by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least:

The minimum decay time for the movement of irradiated fuel in the reactor pressure vessel as determined by the SFP Integrated Decay Heat Management (IDHM) Program*, and shall not be less than 24 hours**.

*The IDHM program will establish the minimum in-vessel decay time needed to assure the SFP limits of 149°F with two available heat exchangers and 180°F with only one heat exchanger prior to the start of each specific Salem refueling outage.

**The current radiological design bases analysis for the Fuel Handling Accident (FHA) is based on a minimum decay time of 24 hours prior to movement of irradiated fuel assemblies within the reactor vessel. Therefore, the decay time for movement of fuel ~~IDHM Program result~~ cannot be less than 24 hours.

~~a. 100 hours - Applicable through year 2010.~~
~~b. 168 hours~~

APPLICABILITY: Mode 6

~~Specification 3.9.3.a - From October 15th through May 15th, during movement of irradiated fuel in the reactor pressure vessel.~~

~~Specification 3.9.3.b - From May 16th through October 14th, during movement of irradiated fuel in the reactor pressure vessel.~~

ACTION:

With the reactor subcritical for less than the required time, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical as required by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by PSEG in this document. Any other statements in this submittal are provided for information only purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Jamie Mallon at (610) 765-5507.

Regulatory Commitment	Due Date/Event
PSEG commits to use the IDHM Program to calculate that the SFP temperature will not exceed 149°F following full core offload, using one heat exchanger per SFP (two heat exchangers total) and to provide to the Operations staff the required Component Cooling Water temperature to achieve such results.	Concurrent with implementation of the amendment
PSEG commits to use the IDHM Program to calculate that the SFP temperature will not exceed 180°F following full core offload with one heat exchanger available for both SFP's and to provide to the Operations staff the required Component Cooling Water temperature to achieve such results.	Concurrent with implementation of the amendment

PROPOSED CHANGES TO TS BASES PAGES

The following Technical Specifications Bases for Salem Unit 1 and Unit 2, Facility Operating License No. DPR-70 and DPR-75, are affected by this change request:

Salem Unit 1

Technical Specification	Page
B 3/4.9.3	B 3/4.9.1b and 1c

Salem Unit 2

Technical Specification	Page
B 3/4.9.3	B 3/4.9.1b and 1c

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, boration to restore the concentration must be initiated immediately. In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

The Surveillance Requirement (SR) ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal, the fuel storage pool and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to reconnecting portions of the refueling canal, the fuel storage pool or the refueling cavity to the RCS, this SR must be met per SR 4.0.4. If any dilution activity has occurred while the cavity or canal was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. A minimum frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The frequency is based on operating experience, which has shown 72 hours to be adequate.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. ~~The 100-hour decay time is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in Reg. Guide 1.183.~~

The minimum decay time for the movement of fuel as determined by the SFP Integrated Decay Heat Management (IDHM) Program will establish the minimum in-vessel decay time needed to assure the SFP limits of 149°F with two available heat exchangers and 180°F with only one heat exchanger prior to the start of each specific Salem refueling outage.

The 24 hour minimum restriction prior to movement of irradiated fuel assemblies within the reactor vessel is based on the current radiological design bases analysis for the Fuel Handling Accident (FHA), and is consistent with the assumptions of the Alternative Source Term described in Reg. Guide 1.183.

~~The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. Delaware River water average temperature between October 15th and May 15th is determined from historical data taken over 30 years. The use of 30 years of data to select maximum temperature is consistent with Reg. Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants".~~

~~A core offload has the potential to occur during both applicability time frames. In order not to exceed the analyzed Spent Fuel Pool cooling capability to maintain the water temperature below 180°F, two decay time limits are provided. In addition, PSEG has developed and implemented a Spent Fuel Pool Integrated Decay Heat Management Program as part of the Salem Outage Risk Assessment. This program requires a pre-outage assessment of the Spent Fuel Pool heat loads and heatup rates to assure available Spent Fuel Pool cooling capability prior to offloading fuel.~~

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

During movement of irradiated fuel assemblies within containment the requirements for containment building penetration closure capability and OPERABILITY ensure that a release of fission product radioactivity within containment will not exceed the guidelines and dose calculations described in Reg. Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors. In MODE 6, the potential for containment pressurization as a result of an accident is not likely. Therefore, the requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements during movement of irradiated fuel assemblies within containment are referred to as "containment closure" rather than containment OPERABILITY. For the containment to be OPERABLE, CONTAINMENT INTEGRITY must be maintained. Containment closure means that all potential containment atmosphere release paths are closed or capable of being closed. Closure restrictions include the administrative controls to allow the opening of both airlock doors and the equipment hatch during fuel movement provided that: 1) the equipment inside door or an equivalent closure device installed is capable of being closed with four bolts within 1 hour by a designated personnel; 2) the airlock door is capable of being closed within 1 hour by a designated personnel, 3) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 4) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

Administrative requirements are established for the responsibilities and appropriate actions of the designated personnel in the event of a Fuel Handling Accident inside containment. These requirements include the responsibility to be able to communicate with the control room, to ensure that the equipment hatch is capable of being closed, and to close the equipment hatch and personnel airlocks within 1 hour in the event of a fuel handling accident inside containment. These administrative controls ensure containment closure will be established in accordance with and not to exceed the dose calculations performed using guidelines of Regulatory Guide 1.183.

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Attachment 5

PSEG Calculation S-C-SF-MEE-1679, Revision 1

SFP Cooling System Capability