PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

MAR 0 8 2007

10 CFR 50.90

LR-N07-0023

United States. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001
Gentlemen:

SALEM GENERATING STATION – UNIT 1 and UNIT 2
FACILITY OPERATING LICENSE NOS. DPR 70 and DPR-75
NRC DOCKET NOS. 50-272 and 50-311

Subject: **RESPONSE TO RAIs ON LCR S06-07**
REFUELING OPERATIONS –DECAY TIME
SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2

References: (1) Letter from PSEG to NRC: "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", dated August 4, 2006

(2) 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

(3) 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

(4) 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

A001

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) previously submitted License Change Request (LCR) S06-07 to amend the Technical Specifications (TS) for the Salem Nuclear Generating Station, Units 1 and 2 (Reference 1). LCR S06-07 proposed to revise the requirements for Fuel Decay Time prior to commencing movement of irradiated fuel. The proposed change revised TS 3/4.9.3 "Decay Time" 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. The proposed change also revised TS 3/4.9.3 to include a limitation on Fuel Decay Time based on the Fuel Handling Accident (FHA) analyses.

The NRC staff provided PSEG a Request for Additional Information (RAI) on LCR S06-07; this request included background material and four specific questions. On January 19th, 2007, PSEG and the NRC staff discussed the RAI via teleconference to provide additional clarification. Attachment 1 to this submittal includes the RAI and the PSEG response. Attachment 2 provides updated information on the IDHM Program. Attachment 3 provides additional changes to the TS. Attachment 4 provides additional changes to the TS Bases.

In accordance with 10CFR50.91(b)(1), a copy of this letter has been sent to the State of New Jersey.

PSEG has evaluated the additional information provided in Attachment 1 in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined there is no impact to the no significant hazards consideration provided in Reference 1. There is also no impact to the 10 CFR 51.22(c)(9) environmental assessment provided in Reference 1.

If you have any questions or require additional information, please do not hesitate to contact Mr. Jamie Mallon at (610) 765-5507.

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

Executed on 3/8/07
(Date)

Sincerely,

Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachments 4

C Mr. S. Collins, Administrator – Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Ennis, Project Manager - Salem Unit 1 and Unit 2
U. S. Nuclear Regulatory Commission
Mail Stop 08B1
Washington, DC 20555-0001

USNRC Senior Resident Inspector – Salem Unit 1 and Unit 2 (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
PO Box 415
Trenton, New Jersey 08625

REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED LICENSE AMENDMENT

REFUELING OPERATIONS - DECAY TIME

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

By letter dated August 4, 2006, PSEG Nuclear LLC (the licensee) submitted an amendment request for Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem). The proposed amendment would revise the Technical Specifications (TSs) to allow the movement of irradiated fuel inside containment to commence at 24 hours after shutdown or at the decay time calculated using the licensee's spent fuel pool (SFP) integrated decay heat management (IDHM) program, whichever is later.

The Nuclear Regulatory Commission (NRC) staff has reviewed the information the licensee provided that supports the proposed amendment and would like to discuss the following issues to clarify the submittal.

Background

Attachment 5 to your letter dated August 4, 2006 (Calculation S-C-SF-MEE-1679, Revision 1), evaluates the cooling capability of the spent fuel pool cooling system (SFPCS). Section 4.1 of Attachment 1 to your letter states, "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 [component cooling water] outlet temperature is less than or equal 71° F." The NRC staff review noted the following issues in review of the calculation regarding whether an 85-hour delay time is conservative for the conditions stated:

a) Tables 3 and 4 of the calculation show switching of SFCS heat exchanger alignment many times at various and predictable time intervals to remove decay heat after full core offload. In Table 3 for normal two heat exchanger operations, the SFCS is switched from normal operation to parallel heat exchanger operation back and forth to maintain both SPFs below 149°F. In Table 4 for one heat exchanger operation, the SFCS is switched from cross-connect heat exchanger operation on one SFP to normal operations on the other SFP back and forth to maintain both SFPs below 180°F. The "Time to Switch HX" time intervals early in the sequences shown in Tables 3 and 4 seem short, especially in Table 4.

   Considering the time required to change the SFP cooling alignment, the NRC staff is concerned whether the heat exchanger switching operations can be performed as listed in Tables 3 and 4 per procedures without exceeding SFP temperature limits of 149°F and 180°F as specified in the Updated Final Safety Analysis Report (UFSAR).

b) Where referring to the stated conservatisms of the calculation, Section 4.1 of Attachment 1 to your letter states, "These inherent conservatisms are of sufficient magnitude to account for any foreseeable changes in river water temperature or other non-conservative assumptions." The NRC staff raises the following concerns regarding this statement:

1) From Figure 9.1-5 of the UFSAR, it appears that there would not be appreciable natural circulation cooling from the reactor vessel to the SFP while residual heat removal cooling of the reactor vessel with all the fuel elements removed.

2) Section 4.1 of Attachment 1 (itemized conservatism 1) states, in part, that: "Consequently, if the pool reaches 180°F, evaporative cooling plus makeup heating removes approximately 9% of the peak heat..." This does not seem to be a conservatism because, as is stated in your letter and the calculation, evaporative cooling is considered in the one heat exchanger case.

3) Calculation S-C-SF-MEE-1679 computes bulk SFP temperature (assumes instantaneous mixing) and computes heat exchanger switching operations and minimum decay time using the UFSAR temperature limits of 149°F and 180°F. If SFP temperature gradients exist, this does not seem conservative.

4) Calculation S-C-SF-MEE-1679, Revision 1, takes credit for the volume of the transfer pool. Revision 0 did not take credit for the transfer pool. Adding the transfer pool increases the available heat capacity by more than 50% per Section 3.2.14 of the calculation. It is questionable whether crediting the volume of the transfer pool in a calculation that uses bulk temperature (instantaneous mixing) while using the UFSAR temperature limits of 149°F and 180°F is conservative.

<u>Questions</u>

Based on the above issues, the NRC staff requests the following additional information:

1) When determining the minimum decay time, how does the IDHM program account for the time needed to switch the heat exchanger alignment of the SFP cooling system back and forth similar to what is required in Calculation S-C-SF-MEE-1679, Revision 1?

RESPONSE

The IDHM Program pre-outage assessment does not permit parallel operation (the back and forth switching of both heat exchangers to one SFP). The cross-tie operation of the SFP heat exchangers (the switching of one available heat exchanger between both SFPs) is a provision that is included only for the situation where a heat exchanger is lost following core off-load. Procedurally (S1(2).OP-IO.ZZ-0007, included in Attachment 2 to this submittal) both heat exchangers are required to be operable prior to core off-load. In addition to the procedural controls, PSEG has committed to operate in this manner. The following commitments, made originally in PSEG letter dated October 2, 2002 (Reference 3), and re-affirmed in Amendment 251/232 SER (Reference 4), are restated and refined below:

> As part of implementation of the requested amendment, PSEG commits to using the IDHM program calculation methodology prior to each Salem refueling outage to:
>
> - Calculate that the SFP temperature will not exceed 149°F following core offload, using one and only one heat exchanger for each SFP and to provide to the Operations staff the required parameters to achieve such results.
>
> - Calculate that the SFP temperature will not exceed 180° F following core offload with one heat exchanger available for both SFP's and to provide to the Operations staff the required parameters to achieve such results.
>
> In addition, as part of implementation of the requested amendment, prior to initiating core offload, PSEG commits to:
>
> 1. Ensuring the availability of both SFP heat exchangers, each with an available spent fuel pit pump, to support spent fuel cooling for core offload; and
>
> 2. Verifying that actual CCW supply temperatures validate the IDHM calculation input requirements.

The above methodology and restrictions are consistent with the IDHM pre-outage assessments performed per Calculations S-C-SF-MDC-1800 (Decay Heat-up Rates and Curves -- Unit 2) and S-C-SF-MDC-1810 (Decay Heat-up Rates and Curves -- Unit 1). These calculations demonstrate that the 149°F SFP temperature limit would not be exceeded with the "hot" pool (pool with core offload) aligned to one and only one SFP heat exchanger (SFHX). The CCW temperature limit is determined for the scheduled offload start time and the current TS limit of 100 hours. These calculations have shown that the required CCW temperature limit is well above the achievable CCW temperature based on the river temperatures at the respective time of year. Utilizing the proposed changes of this LCR, the minimum offload start time will be determined as a function of CCW temperature.

S-C-SF-MEE-1679, Revision 1 (SFP Cooling Capability with Core Offload Starting 85-Hours after Shutdown), which was submitted as Attachment 5 of Reference 1, was prepared to be a bounding analysis of SFP cooling capabilities. It followed the same basic methodology as Revision 0 that had been previously submitted via Reference 3 in support of Amendments 251/232 in 2002 (Reference 4). It relies upon switching heat exchangers back and forth (parallel operation) in order to maintain pool temperatures below 149°F. With the bounding conservative assumptions of S-C-SF-MEE-1679, hot-pool temperature would exceed 149°F using one-and-only one heat exchanger per pool. Comparisons of assumptions between S-C-SF-MEE-1679 and S-C-SF-MDC -1800 (or MDC-1810) are shown in Table 1 below.

Table 1 - Comparison of SFP Cooling Calculation Assumptions

	S-C-SF-MEE-1679	**S-C-SF-MDC-1800/1810**
Purpose	Conservative bounding evaluation for core offload starting 85 hours after shutdown	Realistic outage assessment based upon outage planning schedules and a range of anticipated conditions.
Core Decay-Heat Source	NRC Branch Technical Position ASB 9-2 decay heat estimates with 100% power fuel burn-up throughout the operating cycle	BTP ASB 9-2 methodology with actual element-by-element burn-up data supplied by PSEG Fuels division
Background Pool Heat	Assumes spent fuel in every rack (i.e. full pool)	Uses actual prior offload fuel data
Surface Evaporation	Not considered except for the case where only one HX is available	Uses computer code benchmarked against data from an actual SFP heat-up
Heat Released to Pool Structure	Not considered	Uses computer code benchmarked against data from actual SFP heat-up
Time Off-Load Begins	85-hours	Actual refueling schedule and the 100 Hour TS limit **(Note 1)**

Note 1: A sensitivity study at 85 hours using the IDHM Program (S-C-SF-MDC-1800/1810) determined that the SFP remained below 149°F.

However, since PSEG committed to the IDHM program to assure bulk pool temperatures will be maintained below 149°F using one-and-only one SFP heat exchanger per pool, S-C-SF-MEE-1679 has served its analytical purpose; S-C-SF-MDC-1800 and MDC-1810 provide the appropriate calculation methodology and restrictions consistent with the IDHM Program. S-C-SF-MDC-1810 is provided in Attachment 2 of this submittal, as typical for both units.

While providing more realistic results than S-C-SF-MEE-1679, S-C-SF-MDC-1800 and 1810 have been consistently observed to be conservative when compared to actual bulk pool temperatures during outages. This is reasonable because the BTP ASB 9-2 methodology is conservative even with actual burn-up rates and core decay heat as dominant factors in the calculation. Consequently, S-C-SF-MEE-1679 is replaced for this LCR by the IDHM program calculations S-C-SF-MDC -1800 and MDC-1810.

Based on the proceeding, PSEG will clarify the wording in the TS 3.9.3 (Salem Units 1 and 2) to state:

> *"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 (one and only one heat exchanger per SFP) 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."*

2) How much time do the operators need to align the SFCS for the various configurations required in Tables 3 and 4 of Calculation S-C-SF-MEE-1679, Revision 1?

RESPONSE

The typical time to perform the operational re-alignment of a SFP heat exchanger to the opposite unit is approximately one hour. However, as noted above, the pre-outage assessment will not include the need for any realignment to meet the SFP temperature limits. The only time this would be required is (1) if a heat exchanger becomes unavailable post core offload, or (2) as an operational choice to further reduce the SFP temperature below 149°F.

3) The "Time to Switch HX" column in both Tables 3 and 4 of calculation S-C-SF-MEE-1679, Revision 1 appears to be dependent on the "Tube Flow" values in Table 2, which are 2500 gallons per minute (gpm) and 1500 gpm. How accurate are these two flow rates as compared to actual SFPCS flow rates? Does the IDHM use actual SFPCS flow as determined by either actual measurement or by a flow model?

RESPONSE

The SFP flow rates assumed in S-C-SF-MEE-1679 are based on Calculation S-C-SF-MDC-1780. This calculation documents calculated flows for normal mode (one SFP aligned to one SFHX) and parallel mode (one SFP aligned to two SFHX) system

alignments; it also documents a measured startup test flow for the normal mode system alignment. The normal mode flow of 2500 gpm and parallel mode flow of 1500 gpm per SFP Heat Exchanger were conservatively established based on higher measured and calculated flows, and assumed in S-C-SF-MEE-1679. The IDHM pre-outage calculations likewise assume 2500 gpm for the normal mode system alignment based on S-C-SF-MDC-1780; as stated previously, the pre-outage calculations do not permit parallel mode operation.

The SFP Pumps undergo quarterly surveillance testing. Flow is set to the test value of 2460-2520 gpm, depending on the pump, by throttling a normally full open valve. As such, the actual flow with this valve full open will be higher. This validates the assumed 2500 gpm flow for the normal mode system alignment. Also, the surveillance testing finds that the SFP pumps are operating on or near the design performance curve, i.e. with essentially no degradation. This supports the calculated flows for both the normal and parallel mode system alignments documented in S-C-SF-MDC-1780, which were based on the design performance curve.

Testing performed to benchmark the CROSSTIE computer program (provided in Reference 3) included switching the Unit 2 SFHX between the Unit 1 and Unit 2 SFPs. With the Unit 1 SFP aligned to the Unit 2 SFHX, the flow was found to be 2460 gpm; this is less than what would have been expected if the Unit 1 SFP was aligned directly to its SFHX. This further validates the assumed 2500 gpm flow for the normal mode system alignment. With the Unit 2 SFP aligned to the Unit 2 SFHX, however, the flow was found to be only 2040 gpm. Based on the data provided above, this value is considered to be an anomaly. Surveillance testing has consistently demonstrated the ability to achieve 2500 gpm.

4) The amendment request proposes to revise TS Limiting Condition for Operation (LCO) 3.9.3 to reference the IDHM program as the means to determine the LCO decay time. Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(c)(5) requires, in part, that the TSs contain administrative controls relating to procedures necessary to assure operation of the facility in a safe manner. Consistent with these requirements, the IDHM program should be added to TS Section 6.0, "Administrative Controls." Please submit your IDHM program for review and revise TS Section 6.0 accordingly.

RESPONSE

The IDHM Program description will be added to Section 6.0 of TS, as described below:

6.8.4.j[1] *INTEGRATED DECAY HEAT MANAGEMENT (IDHM) PROGRAM*

The IDHM Program is part of the Salem Outage Risk Assessment Management (ORAM) Program. The IDHM Program requires a pre-outage assessment of the

[1] Another change to Section 6.8.4 is currently pending for Salem Unit 2 – this may impact the numbering of the IDHM Program section.

SFP heat loads and heat-up rates to assure available SFP cooling capability prior to offloading fuel. The program has the following requirements:

a. *The SFP temperature will not exceed 149°F following core offload, using one and only one heat exchanger for each SFP. The Operations staff will be provided the required parameters to achieve such results.*

b. *The SFP temperature will not exceed 180°F following core offload with one heat exchanger available for both SFP's. The Operations staff will be provided the required parameters to achieve such results.*

c. *The availability of both SFP heat exchangers, each with an available spent fuel pit pump, to support spent fuel cooling for a core offload.*

d. *The verification that actual CCW supply temperatures validate the IDHM calculation input requirements.*

PSEG previously submitted the IDHM Program, which included the Crosstie computer program, to the NRC Staff for review (Reference 3). The review of the program was documented in Amendment 251 & 232 SER (Reference 4). The Crosstie computer program has not changed. PSEG is providing the following updated information on the IDHM Program (Attachment 2 to this submittal):

- Calculation S-C-SF-MDC-1810 (typical for both units)
- PSEG Procedure SC.OM-AP.ZZ-0001
- PSEG procedure S1.OP-IO.ZZ-0007

The IDHM Program elements are documented in the following sections of the above procedures:

SC.OM-AP.ZZ.0001	S1.OP-IO.ZZ-0007
5.7	5.2.2
Attachment 5	5.3.1.F
Attachment 6	Attachment 3

Additional IDHM Program Documentation

- Calculation S-C-SF-MDC-1810
- PSEG Procedure SC.OM-AP.ZZ-0001
- PSEG procedure S1(2).OP-IO.ZZ-0007

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
6.8.4.j	6-19d

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
6.8.4.j	6-19b, c

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 (one and only one heat exchanger per SFP) 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.

based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.8.4.j INTEGRATED DECAY HEAT MANAGEMENT (IDHM) PROGRAM

The IDHM Program is part of the Salem Outage Risk Assessment Management (ORAM) Program. The IDHM Program requires a pre-outage assessment of the SFP heat loads and heat-up rates to assure available SFP cooling capability prior to offloading fuel. The Program has the following requirements:

a. The SFP temperature will not exceed 149°F following core offload, using one and only one heat exchanger for each SFP. The Operations staff will be provided the required parameters to achieve such results.

b. The SFP temperature will not exceed 180°F following core offload with one heat exchanger available for both SFP's. The Operations staff will be provided the required parameters to achieve such results.

c. The availability of both SFP heat exchangers, each with an available spent fuel pit pump, to support spent fuel cooling for a core offload.

d. The verification that actual CCW supply temperatures validate the IDHM calculation input requirements.

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 (one and only one heat exchanger per SFP) 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.

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iondine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.4.h Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of the census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.4.j INTEGRATED DECAY HEAT MANAGEMENT (IDHM) PROGRAM

Note: changes to this page are currently pending – LCR S06-01 (TAC MD1193)

The IDHM Program is part of the Salem Outage Risk Assessment Management (ORAM) Program. The IDHM Program requires a pre-outage assessment of the SFP heat loads and heat-up rates to assure available SFP cooling capability prior to offloading fuel. The Program has the following requirements:

a. The SFP temperature will not exceed 149°F following core offload, using one and only one heat exchanger for each SFP. The Operations staff will be provided the required parameters to achieve such results.

b. The SFP temperature will not exceed 180°F following core offload with one heat exchanger available for both SFP's. The Operations staff will be provided the required parameters to achieve such results.

c. The availability of both SFP heat exchangers, each with an available spent fuel pit pump, to support spent fuel cooling for a core offload.

d. The verification that actual CCW supply temperatures validate the IDHM calculation input requirements.

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.~~

Insert 1

INSERT 1

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 (one and one only per SFP) and 180°F with only one heat exchanger prior to the start of each specific Salem refueling outage.

The scenario of only one available heat exchanger is an abnormal operating situation, which may require switching the available heat exchanger between SFPs. Both heat exchangers are required to be available prior to the start of fuel offload. The crosstie evaluation demonstrates satisfactory operation for the abnormal situation due of an unavailable heat exchanger post core-offload.

In addition, parallel operation (two heat exchangers aligned to one pool) is an abnormal operating situation, to further reduce the SFP temperature below the 149 degrees limit (e.g., for habitability reasons). This may be an operational choice, post-offload.

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.

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.~~

Insert 1

INSERT 1

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