Robert Braun
Site Vice President-Salem

PSEG Nuclear LLC
Salem Nuclear Generating Station
P.O. Box 236, M/C S-05, Hancocks Bridge, NJ 08038
tel: 856.339.1998 fax: 856.339.2956
email: Robert.Braun@pseg.com

PSEG
Nuclear L.L.C.

JUN 17 2008

10 CFR 50.90

LR-N08-0135

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

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: **SUPPLEMENT (REDUCED SCOPE) - REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS, REFUELING OPERATIONS – DECAY TIME**
LICENSE AMENDMENT REQUEST (LAR) S08-01

References: (1) Letter from PSEG to NRC: "Request for Changes to Technical Specifications, Refueling Operations –Decay Time, LAR S08-01, Salem Nuclear Generating Station, Unit 1 and Unit 2, Facility Operating License DPR-70 and DPR-75, Docket Nos. 50-272 and 50-311", dated March 11, 2008

In Reference 1, PSEG Nuclear LLC (PSEG) submitted License Amendment Request (LAR) S08-01, proposing revisions to the requirements for fuel decay time prior to commencing movement of irradiated fuel. TS 3/4.9.3 "Decay Time" would be (1) revised to allow fuel movement to commence at 80 hours after the reactor is subcritical between October 15th and May 15th, and (2) relocated to the Salem UFSAR, or Technical Requirements Manual (TRM). Currently, TS 3/4.9.3 requires a fuel decay time of 100 hours prior to fuel movement between October 15th and May 15th.

Subsequent to the submittal of Reference 1, PSEG and the NRC staff have discussed the LAR to provide additional clarification. Based on these discussions PSEG proposes to limit the scope of the LAR to: revise the TS to allow fuel movement to commence at 80 hours after the reactor is subcritical between October 15th and May 15th (Item 1 above). The previous request to relocate the TS to the UFSAR or TRM (Item 2 above) is no longer being proposed.

A001
NRR

Specifically, in Attachment 1 of Reference 1, the request to relocate in Section 2 is no longer being proposed, along with the supporting discussion and analysis in Sections 3.2 and 4.2. Accordingly, Section 5.0, the 10 CFR 50.91(a) no significant hazards considerations evaluation, has been revised and is attached to this submittal (Attachment 1). The revised limited scope of this amendment request continues to meet the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

Attachment 2 provides the existing TS pages marked-up to show the proposed changes based on the revised limited scope. Attachment 3 provides the existing TS Bases pages marked-up to reflect the associated changes to the TS (for information only).

In accordance with 10 CFR 50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG requests approval of the proposed License Amendment by September 30, 2008 to be implemented within 30 days, to support Salem Unit 1 refueling outage 1R19.

If you have any questions or require additional information, please do not hesitate to contact Mr. Jeff Keenan at (856) 339-5429.

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

Executed on 6/17/08
(Date)

Sincerely,

Robert C. Braun
Site Vice President
Salem Generating Station

Attachments: 3

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

Mr. P. Mulligan
Bureau of Nuclear Engineering
PO Box 415
Trenton, New Jersey 08625

SUPPLEMENT (REDUCED SCOPE) - REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS, REFUELING OPERATIONS – FUEL DECAY TIME LICENSE AMENDMENT REQUEST (LAR) S08-01

PSEG Nuclear LLC (PSEG) submitted License Amendment Request (LAR) S08-01[1], proposing revisions to the requirements for fuel decay time prior to commencing movement of irradiated fuel. TS 3/4.9.3 "Decay Time" would be (1) revised to allow fuel movement to commence at 80 hours after the reactor is subcritical between October 15th and May 15th, and (2) relocated to the Salem UFSAR, or Technical Requirements Manual (TRM). Currently, TS 3/4.9.3 requires a fuel decay time of 100 hours prior to fuel movement between October 15th and May 15th.

Subsequent to the submittal of LAR S08-01, PSEG and the NRC staff have discussed the LAR to provide additional clarification. Based on these discussions PSEG proposes to limit the scope of the LAR to: revise the TS to allow fuel movement to commence at 80 hours after the reactor is subcritical between October 15th and May 15th (Item 1 above). The previous request to relocate the TS to the UFSAR or TRM (Item 2 above) is no longer being proposed.

Specifically, in Attachment 1 of LAR S08-01, the request to relocate in Section 2 is no longer being proposed, along with the supporting discussion and analysis in Sections 3.2 and 4.2. Accordingly, Section 5.0, the 10 CFR 50.91(a) no significant hazards considerations evaluation, has been revised (below). The revised limited scope of this amendment request continues to meet the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

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] Letter from PSEG to NRC: "Request for Changes to Technical Specifications, Refueling Operations – Decay Time, LAR S08-01, Salem Nuclear Generating Station, Unit 1 and Unit 2, Facility Operating License DPR-70 and DPR-75, Docket Nos. 50-272 and 50-311", dated March 11, 2008

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 (SFP) in less time after subcriticality than currently allowed by the Technical Specifications. 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 accident previously evaluated that is associated with the proposed license amendment is the fuel handling accident. Allowing the fuel to be offloaded in less time after subcriticality using actual heat loads 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 (FHA). The time frame in which the fuel assemblies are moved has been evaluated against the 10CFR50.67 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 significantly increase the probability of occurrence or the consequences of accidents previously evaluated.

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 which affects the radionuclide makeup 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; no new accidents are introduced. 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 demonstrates that the existing Spent Fuel Pool cooling system and associated systems under worst-case circumstances would maintain licensing limits and the integrity of the Spent Fuel Pool.

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 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 heat-up calculations performed prior to the refueling outage, as part of the Integrated Decay Heat Management (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

10 CFR 50 Appendix A, General Design Criteria 5--Sharing of structures, systems, and components.

GDC 5 requires that structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

In the unlikely event that a SFP heat exchanger (HX) becomes unavailable after core off-load begins, the sharing of one SFP heat exchanger between units has been adequately evaluated. In the worst case scenario, Operations has adequate time to cross-connect the available heat exchanger to the refuel pool before 180°F is reached; the time to make the cross-connect has been appropriately addressed with procedural controls. When the HX is aligned to the non-refuel pool, temperature is rapidly brought down, so the time needed to swap back again to the non-refuel pool is much greater. In addition, Operations has the expected pool heat-up rates prior to fuel off-load; they monitor actual heat-up rates, so they can anticipate required actions.

10 CFR 50 Appendix A, General Design Criteria 61--Fuel storage and handling and radioactivity control.

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

The changes proposed by the amendment request do not reduce the existing UFSAR requirements for meeting GDC 61. The heat removal capability of the SFP cooling system is maintained for normal, abnormal and accident conditions.

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 is consistent with the current requirements of GDC 19 for the Fuel Handling Accident.

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.

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 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 SFP temperature limits are not exceeded.

The doses are less than the TEDE criteria set forth in RG 1.183 and are a small fraction of the dose criteria in 10 CFR 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.

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License DPR-70 and DPR-75 are affected by this change request:

DPR-70, Salem Unit 1

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

DPR-75, Salem Unit 2

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

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least:.

a. ~~100~~ 80 hours ~~Applicable through year 2010.~~
b. 168 hours

APPLICABILITY: 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:

a. ~~100*~~ 80 hours ~~Applicable through year 2010~~
b. 168 hours

APPLICABILITY: Specification 3.9.3.a - From October 15^{th} through May 15^{th}, during movement of irradiated fuel in the reactor pressure vessel.

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

~~* For Salem Unit 2 refueling outage 2R16 only, the required subcritical time is 86 hours~~

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.

PROPOSED CHANGES TO TS BASES PAGES (INFORMATION ONLY)

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

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~~80-hour decay time (LAR S08-01) 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.

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~~80-hour decay time (LAR S08-01) 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 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 heat-up rates to assure available Spent Fuel Pool cooling capability prior to offloading fuel.

~~For Salem Unit 2 refuel outage 2R16, a specific analysis was done to allow for an hour decay time (reference LAR S07-06). The impact of the increase in heat load in the SFP (due to the reduced decay time) on the Fuel Handling Accident and SFP cooling requirements was evaluated. The dose remains less than the TEDE criteria set forth in RG 1.183 and is a small fraction of the dose criteria in 10CFR 50.67. The SFP cooling system remains capable of (1) maintaining both Salem pools below 149°F with two SFP heat exchangers available and (2) maintaining both pools below 180°F with only one heat exchanger available. This capability meets the requirements of UFSAR Chapter 9.1.3.1.~~

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 Plants. 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 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 doors are capable of being closed within 1 hour by 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.