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PSEG
Nuclear LLC

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U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
LICENSE AMENDMENT REQUEST
REVISION TO TECHNICAL SPECIFICATIONS 3/4.9.4, 3/4.9.5, 3/4.9.6 AND 3/4.9.7
REFUELING OPERATIONS

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC hereby requests a revision to the Technical Specifications (TS) for the Hope Creek Generating Station. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed amendment will relocate portions of Technical Specification Section 3/4.9, "Refueling Operations," and associated Bases from the Technical Specifications to the Hope Creek Updated Final Safety Analysis Report (UFSAR).

The proposed changes are consistent with NUREG-1433, Standard Technical Specifications, General Electric Plants, BWR/4, Revision 1, dated April 1995, and with the NRC's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (58 FR 39132), dated July 22, 1993. PSEG Nuclear 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. The basis for the requested change is provided in Attachment 1 to this letter. A 10CFR50.92 evaluation, with a determination of no significant hazards consideration, is provided in Attachment 2. The marked-up Technical Specification pages affected by the proposed changes are provided in Attachment 3.

PSEG Nuclear requests approval of the proposed License Amendment by December 15, 2001 to be implemented within 60 days.

A001

Should you have any questions regarding this request, please contact Mr. Paul Duke at 856-339-1466.

Sincerely,

David F. Garchow
Vice President – Operations

Affidavit
Attachments (3)

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NBU RM (N64)
Microfilm Copy
Files 1.2.1 (Hope Creek)
2.3 (LCR H01-01)

REF: LRN-01-0092
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STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

David F. Garchow, being duly sworn according to law deposes and says:

I am Vice President – Operations of PSEG Nuclear LLC, and as such, I find the matters set forth in the above referenced letter, concerning Hope Creek Generating Station, Unit 1, are true to the best of my knowledge, information and belief.

David F. Garchow

Subscribed and Sworn to before me
this 2 day of April, 2001

Sheri L. Huston
Notary Public of New Jersey

SHERI L. HUSTON
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 12/08/2003

My Commission expires on

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
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REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

BASIS FOR REQUESTED CHANGE

PSEG Nuclear is requesting a change to the Hope Creek Technical Specifications (TS) that will relocate portions of TS Section 3/4.9, Refueling Operations, to the Hope Creek Updated Final Safety Analysis Report (UFSAR) which is controlled under the requirements of 10CFR50.59. The requested change is consistent with NUREG-1433, Standard Technical Specifications, General Electric Plants, BWR/4, Revision 1, dated April 1995, and with the NRC's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (58 FR 39132), dated July 22, 1993.

REQUESTED CHANGE, PURPOSE AND BACKGROUND:

The requested change will relocate the following Technical Specifications and their associated Bases to the Hope Creek UFSAR:

3/4.9.4, Decay Time

3/4.9.5, Communications

3/4.9.6, Refueling Platform

3/4.9.7, Crane Travel - Spent Fuel Pool

The NRC described the purpose of Technical Specifications in its Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors:

> "...to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by identifying those features that are of controlling importance to safety and establishing on them certain conditions of operation which cannot be changed without prior Commission approval."

10 CFR 50.36 provides four specific criteria to delineate those constraints on design and operation of nuclear power plants that are derived from the plant safety analysis report or probabilistic safety assessment (PSA) information and that shall be included in the Limiting Conditions for Operation (LCOs) in the Technical Specifications. Existing LCOs which do not meet any of the criteria in 10 CFR 50.36 may be proposed for removal from the Technical Specifications and relocation to licensee-controlled documents.

The LCOs proposed for removal and relocation do not meet any of the criteria in 10 CFR 50.36. Their relocation to the Hope Creek UFSAR will provide additional operational flexibility during refueling outages.

JUSTIFICATION OF REQUESTED CHANGES:

10 CFR 50.36 requires that a technical specification limiting condition for operation (LCO) be established for each item meeting one or more of the following criteria:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Existing LCOs which do not meet any of the criteria in 10 CFR 50.36 may be proposed for removal from the Technical Specifications and relocation to licensee-controlled documents.

TS 3/4.9.4, DECAY TIME

LCO Statement
The reactor shall be subcritical for at least 24 hours.

Discussion
The TS establishes a minimum time requirement for reactor subcriticality before movement of irradiated fuel in the reactor pressure vessel to ensure sufficient time has elapsed to allow the radioactive decay of short-lived fission products. The 24-hour period for decay following subcriticality will continue to be met for a refueling outage due to procedural controls on operations required before moving irradiated fuel in the reactor pressure vessel (e.g., containment entry, removal of the drywell head, removal of the vessel head, removal of vessel internals). Therefore, the requirement can be relocated from the Technical Specifications.

Comparison to Screening Criteria:

1. The Decay Time specification does not involve installed instrumentation used to detect, or indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The minimum time requirement for reactor subcriticality before movement of irradiated fuel in the reactor pressure vessel is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, procedural controls on operations required before moving irradiated fuel in the reactor pressure vessel ensure the 24-hour decay time following subcriticality will continue to be met for a refueling outage. Therefore, the requirement can be relocated from the Technical Specifications.

3. The Decay Time specification does not involve a system, structure or component which is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Although the risk significance of the Decay Time specification was not directly evaluated in NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment", those refueling operations that were evaluated were found to be non-significant contributors to core damage frequency and offsite releases.

Conclusion:
Consistent with the criteria delineated in 10 CFR 50.36, the Decay Time LCO and Surveillances may be relocated to other licensee-controlled documents outside the Technical Specifications.

TS 3/4.9.5, COMMUNICATIONS

LCO Statement:
Direct communication shall be maintained between the control room and refueling floor personnel.

Discussion:
Communications between the control room and refueling floor is maintained to ensure that refueling personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and refueling floor personnel (such as the insertion of a control rod prior to loading fuel). However, the refueling system design accident or transient response does not take credit for communications and is designed to ensure safe refueling operations.

Comparison to Screening Criteria:
1. Communications between the control room and refueling floor personnel are not used to detect, or indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Communications between the control room and refueling floor personnel are not an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Communications between the control room and refueling floor personnel are not part of the primary success path to mitigate a design basis accident or transient.

4. As discussed in Sections 3.5 and 6.9 and summarized in Table 4-1 (item 286) of NEDO-31466, the loss of communications was found to be a non-significant contributor to core damage frequency and offsite releases. PSEG Nuclear has reviewed this evaluation, considers it applicable to Hope Creek, and concurs with the assessment.

Conclusion:
Consistent with the criteria delineated in 10 CFR 50.36, the Communications LCO and Surveillances may be relocated to other licensee-controlled documents outside the Technical Specifications.

TS 3/4.9.6, REFUELING PLATFORM

LCO Statement:
The refueling platform shall be OPERABLE with the main hoist to be used for handling fuel assemblies or control rods within the reactor pressure vessel and the frame-mounted or monorail-mounted auxiliary hoists to be used for handling control rods within the reactor pressure vessel.

Discussion:
Operability of the refueling platform equipment (cranes, main hoist and auxiliary hoist) ensures that: (1) only the main hoist of the refueling platform will be used to handle fuel within the reactor pressure vessel; (2) hoists have sufficient load capacity for handling fuel assemblies and/or control rods; and (3) the core internals and pressure vessel are protected from excessive lifting force in the event that they are inadvertently engaged during lifting operations. Although the interlocks designed to provide the above capabilities can prevent damage to the refueling platform equipment and core internals, they are not assumed to function to mitigate the consequences of a design basis accident. Technical Specification limits on reactor mode switch position (TS 3/4.9.1) remain in place to reinforce refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radiation.

Comparison to Screening Criteria:
1. The refueling platform and associated instrumentation are not used to detect, or indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The refueling platform and associated instrumentation are not used to monitor a process variable that is an initial condition of a design basis accident or transient.

3. The refueling platform and associated instrumentation are not part of the primary success path to mitigate a design basis accident or transient.

4. As discussed in Sections 3.5 and 6.9 and summarized in Table 4-1 (item 287) of NEDO-31466, the refueling platform and associated instrumentation were found to be a non-significant contributor to core damage frequency and offsite releases. PSEG Nuclear has reviewed this evaluation, considers it applicable to Hope Creek, and concurs with the assessment.

Conclusion:
Consistent with the criteria delineated in 10 CFR 50.36 the Refueling Platform LCO and Surveillances may be relocated to other licensee-controlled documents outside the Technical Specifications.

TS 3/4.9.7, CRANE TRAVEL - SPENT FUEL STORAGE POOL

LCO Statement:
Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks unless handled by a single failure proof handling system.

Discussion:
The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that, in the event the load is dropped, (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. The crane travel requirements are implemented by a combination of crane interlocks and administrative controls on the handling of heavy loads.

Although this TS is intended to support the maximum refueling accident assumption in the design basis accident analysis, crane travel limits are not monitored and controlled during plant operation; they are checked on a periodic basis to assure operability.

Comparison to Screening Criteria:
1. The crane travel limits are not used to detect, or indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The maximum severity assumed for the fuel handling design basis accident analysis is limited by the limits on crane travel. However, these limits are not process variables monitored and controlled by the operator. They are a

combination of interlocks and physical stops and administrative controls. Therefore, Criterion 2 is not satisfied.

3. The crane travel limits are not part of the primary success path to mitigate a design basis accident or transient.

4. While probabilistic risk assessments do not typically review the risks associated with the spent fuel storage pool, design basis analyses indicate the release associated with fuel assembly damage in the spent fuel storage pool due to refueling accidents is significantly lower than the releases evaluated by PRAs.

Conclusion:
Consistent with the criteria delineated in 10 CFR 50.36, the Crane Travel - Spent Fuel Storage Pool LCO and Surveillances may be relocated to other licensee-controlled documents outside the Technical Specifications.

ENVIRONMENTAL EVALUATION

The proposed TS changes were reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Based on the foregoing, PSEG Nuclear concludes that the proposed TS changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement. Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

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10CFR50.92 EVALUATION

PSEG Nuclear has concluded that the proposed changes to the Hope Creek Generating Station (HC) Technical Specifications do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGE

The requested change will relocate portions of Technical Specification Section 3/4.9, "Refueling Operations," and associated Bases from the Technical Specifications to the Hope Creek Updated Final Safety Analysis Report (UFSAR) which is controlled under the requirements of 10CFR50.59.

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in 10CFR50.92. This regulation states that a proposed amendment involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed change has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

1. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The requested amendments will not involve an increase in the probability or consequences of an accident previously evaluated. Relocation of the affected Technical Specification sections and their Bases to the Hope Creek UFSAR will have no affect on the probability that any accident will occur. Additionally, the consequences of an accident will not be impacted because the affected systems and components will continue to be utilized in the same manner as before. No impact on the plant response to accidents will be created.

   Based on the above, the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. *The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

   The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms will be created as a result of the relocation of the affected Technical Specification requirements and their Bases to the Hope Creek UFSAR. Plant operation will not be affected by the proposed amendments and no new failure modes will be created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. *The proposed change does not involve a significant reduction in a margin of safety.*

   The proposed amendments will not involve a reduction in the margin of safety. Relocation of the affected Technical Specification requirements to the Hope Creek UFSAR is consistent with NUREG 1433, Standard Technical Specifications, General Electric Plants, BWR/4, Revision 1, dated April 1995, and with the NRC's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (58 FR 39132), dated July 22, 1993, which encourages utilities to propose amendments consistent with NUREG 1433. The margin of safety is unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed changes do not involve a significant hazards consideration.

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TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

Technical Specification	Page
INDEX	xiv and xx
3/4.9.4	3/4 9-6
3/4.9.5	3/4 9-7
3/4.9.6	3/4 9-8
3/4.9.7	3/4 9-10
B 3/4.9.4	B 3/4 9-1
B 3/4.9.5	B 3/4 9-1
B 3/4.9.6	B 3/4 9-2
B 3/4.9.7	B 3/4 9-2

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION		PAGE

3/4.7 PLANT SYSTEMS

3/4.9.4 DECAY TIME DELETED

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.5 COMMUNICATIONS

DELETED

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.~~

~~APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.~~

~~ACTION:~~

~~When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.~~

3/4.9.6 REFUELING PLATFORM DELETED

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE with the main hoist to be used for handling fuel assemblies or control rods within the reactor pressure vessel and the frame-mounted or monorail-mounted auxiliary hoists to be used for handling control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6. The refueling platform hoists used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1200 + 0, -50 pounds.

b. Demonstrating operation of the overload cutoff on the frame-mounted and monorail-mounted auxiliary hoists when the load exceeds 500 ± 50 pounds.

c. Demonstrating operation of the main hoist uptravel stop when uptravel brings the point where the grapple attaches to the fuel bundle to 6 feet 6 inches, +3, -0 inches below the normal water level.

d. Demonstrating operation of the frame-mounted and monorail-mounted auxiliary hoists' uptravel stops when uptravel brings the point where the grapple attaches to a control rod to 6 feet, +1, -0 feet below the normal water level.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

f. Demonstrating operation of the loaded rod block interlock on the main hoist when the load exceeds 535 ± 50 pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 ± 50 pounds.

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

DELETED

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks unless handled by a single failure proof handling system.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the polar crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Interlocks and physical stops which prevent polar crane main hoist travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

4.9.7.2 The single failure proof handling system shall be visually inspected and verified OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radiation.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. The flux need not be monitored for the first sixteen bundles loaded before a SPIRAL RELOAD or for the last sixteen bundles unloaded during a SPIRAL UNLOAD. In the case of the SPIRAL RELOAD, the sixteen bundles loaded may be different from the bundles scheduled to occupy the bundle locations for the next cycle provided; (i) the cold reactivity of any unscheduled bundle temporarily loaded is individually less than the cold reactivity of the respective bundle scheduled for the subject location, (ii) the uncontrolled k-infinity of the lattice is less than 1.31, and (iii) the bundles are arranged in four two-by-two arrays surrounding an SRM with each array having a minimum of 12 inches between it and an adjacent array.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS minimizes the possibility that fuel will be loaded into a cell without a control rod, although one rod may be withdrawn under control of the reactor mode switch refuel position one-rod-out-interlock.

3/4.9.4 DECAY TIME DELETED

~~The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.~~

3/4.9.5 COMMUNICATIONS DELETED

~~The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.~~

3/4.9.6 REFUELING PLATFORM DELETED

~~The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling fuel assemblies and control rods, with limits placed upon auxiliary hoists' usage, within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling the loads within its permitted usage, (3) the core internals are protected from excessive lifting force in the event that they are inadvertently engaged during lifting operations, (4) the core internals are protected from a fuel bundle or control rod drop with more impact energy than that assumed in the accident analyses, (5) refueling interlocks and rod blocks are initiated to prevent conditions that could result in criticality during refueling operations.~~

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL DELETED

~~The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.~~

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL - SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and (2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.