NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 20, 1998, through December 4, 1998. The last biweekly notice was published on December 2, 1998 (63 FR 66590).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that 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; or (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 basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 15, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland.

Date of amendments request: November 19, 1998.

Description of amendments request: The proposed amendment revises Technical Specification 3.7.6, ''Service Water (SRW) System'' to allow operation of Calvert Cliffs with one SRW plate and frame heat exchanger (PHE) secured for maintenance or other reasons, and removing one containment air cooler (CAC) from service to enable the affected subsystem to remain operable. Specifically, the proposed change adds ''One SRW heat exchanger inoperable'' as a new condition for Limiting Condition for Operation (LCO) 3.7.6. The required actions for the new condition are to secure one CAC within one hour and restore the heat exchanger to operable condition within 7 days, or be in Mode 3 in 6 hours and Mode 5 in 36 hours. This limits the effect of one inoperable PHE to only one containment cooling train made inoperable by the PHE. Consequently, the new action statement introduced in the SRW LCO for an inoperable PHE is similar to the one that already exists in the CAC LCO for one inoperable containment cooling train.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

None of the systems associated with the proposed revision to the Calvert Cliffs Technical Specifications are accident initiators. The Saltwater (SW) and SRW systems are used to mitigate the effects of accidents analyzed in the Updated Final Safety Analysis Report (UFSAR). The SW and SRW Systems provide cooling to safety-related equipment following an accident. The CACs are provided with SRW to remove heat from the Containment in the event of an accident. They support accident mitigation functions; therefore, the proposed modification does not increase the probability of an accident previously evaluated.

The proposed revision will provide greater availability of safety-related equipment during PHE maintenance activities. It ensures that the safety features provided by the SW and SRW, except for the isolated CAC, are maintained, i.e., the availability of safety-related equipment required to mitigate the radiological consequences of an accident described in the UFSAR is enhanced by the flexibility provided by this Technical Specification revision.

Furthermore, the proposed revision will not change, degrade, or prevent actions described or assumed in any accident described in the UFSAR. The proposed activity will not alter any assumptions previously made in evaluating the radiological consequences of any accident described in the UFSAR.

Therefore, the proposed modification does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

None of the systems associated with this modification are identified as accident initiators in the UFSAR. The SW and SRW Systems and the CACs are used to mitigate the effects of accidents analyzed in the UFSAR. None of these functions required of these systems have been changed by the proposed revision to the Technical Specifications. This activity does not modify any system, structure, or component such that it could become accident initiator, as opposed to its current role as an accident mitigator.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The safety design basis for the SW and SRW Systems is the availability of sufficient cooling capacity to ensure continued operation of equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with assumptions used in the accident analysis.

With one SRW subsystem inoperable, the remaining SRW subsystem is adequate to perform the heat removal function. However, the reliability is reduced because a single failure in the operable SRW subsystem could result in loss of SRW function. The proposed change will allow continued operation of some SRW-cooled components while a PHE is being out-of-service. The second SRW subsystem will still be available to perform the SRW function. In addition, the reliability of many diesel generator-backed components will be improved since the second diesel generator will remain operable while in this action statement.

During a design basis accident, a minimum of one containment cooling train (two of the four CACs) and one containment spray train, is required to maintain the containment peak pressure and temperature, below the design limits. Under the existing Technical Specification requirement, with one containment cooling train inoperable, the inoperable containment cooling train must be returned to operable status within seven days. The remaining operable containment spray and cooling units provide iodine removal capabilities and are capable of removing at least 100% of the heat removal needs after an accident. The seven-day completion time was developed taking into account the redundant heat removal capabilities afforded by combinations of the containment spray and cooling systems, and the low probability of a design basis accident occurring during this period. The proposed change to Technical Specification 3.7.6 would allow three CACs to remain operable during maintenance on a PHE, instead of the two that are maintained under the current Technical Specification requirement.

For the above reasons, the margin of safety has been preserved, and in some cases increased, by the proposed revision to the Technical Specifications.

Therefore, this proposed modification does not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: S. Singh Bajwa, Director.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland.

Date of amendments request: November 20, 1998.

Description of amendments request: On September 9, 1996, a final rule amending 10 CFR 50.55a was issued requiring owners to implement, by September 9, 2001, the requirements of the 1992 Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Subsections IWE and IWL, as modified and supplemented by 10 CFR 50.55a. Baltimore Gas and Electric Company (BGE) have developed a program plan to effect the implementation of Subsection IWE and IWL. BGE's submittal requests a license amendment in support of the program plan. One Technical Specification (TS) change requested is an administrative change that removes a TS originally developed from Regulatory Guide (RG) 1.35. Compliance with RG 1.35 is not sufficient to comply with 10 CFR 50.55a, as amended. The other TS changes request the removal from the TSs requirements that are a duplication of 10 CFR 50.55a.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Containment Building is a passive safety structure that prevents the release of radioactive materials to the environment in post-accident conditions. The proposed Technical Specification changes delete requirements of the Technical Specifications that have been made obsolete by the improvements of the Containment Building inspections required by the changes in the regulations. The improved inspections required by the American Society of Mechanical Engineers Code serve to maintain Containment response to accident conditions, by causing the identification and repair of defects in the Containment Buildings.

Relocating existing requirements, eliminating requirements that duplicate regulations, and making administrative improvements provide Technical Specifications that are easier to use. Because existing requirements are controlled by regulation, there is no reduction in commitment and adequate control is still maintained. Likewise, the elimination of requirements that duplicate regulations enhances the usability of the Technical Specifications without reducing commitments. Therefore, the proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The Containment Building is a passive safety structure designed to contain radioactive materials released from the Reactor Coolant System. The performance of the Containment Building is not evaluated as the causal factor in any accident at Calvert Cliffs Nuclear Power Plant. The proposed Technical Specification changes delete requirements of the Technical Specifications that have been made obsolete by the improvements of the Containment Building inspections required by the changes in the regulations. Revising the Technical Specifications, to comply with current regulations and to eliminate duplication of requirements, does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The safety function of the Containment Building is to provide a boundary to the release of radioactive material to the environment during post-accident conditions. The changes to the Technical Specifications incorporate improved inspection techniques and criterial to ensure optimum Containment integrity and, therefore, optimum containment response in the event of an accident resulting in a release of radioactive material from the Reactor Coolant System.

Optimizing containment integrity will result in maintaining the margin of safety allowed by the Containment Buildings. Therefore, the proposed changes will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: S. Singh Bajwa, Director.

CBS Corporation acting through its Westinghouse Electric Company Division (licensee), Westinghouse Test Reactor, Waltz Mill Site, Westmoreland, Pennsylvania, Docket No. 50–22, License No. TR–2.

Date of amendment request: September 28, 1998, supplemented on November 17, 1998.

Description of amendment request: CBS Corporation acting through its Westinghouse Electric Company Division is the licensee for the Westinghouse Test Reactor (WTR) at Waltz Mill, Pennsylvania. The licensee is authorized to only possess the reactor and a decommissioning plan has been approved. The licensee is planning to sell most of its nuclear related facilities to other entities, but will retain the WTR. One of the arrangements made with the purchasers of the other facilities is that the Westinghouse name will be conveyed with these facilities, and because of this arrangement, the licensee requests that the license associated with the Westinghouse Test Reactor be changed to simply CBS Corporation, to eliminate any reference to the name Westinghouse.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards considerations. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). A proposed amendment to a license of a facility 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 or consequences of an accident previously evaluated; or (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 staff agrees with the licensee's no significant hazards consideration determination submitted on November 17, 1998, for the following reason.

This corporate name change does not involve any change in the management, organization, location, facilities equipment, or procedures related to the licensed activities under the WTR license. The employees responsible for the licensed WTR facility will still be responsible, either directly through the CBS Corporation or through contractual arrangements for which CBS Corporation is ultimately responsible, notwithstanding the new name of the licensee.

Based on a review of the licensee's analysis, and on the staff's analysis detailed above, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for license: Lisa A. Campagna, Assistant General Counsel, Law Department, CBS Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

NRC Project Director: Seymour H. Weiss.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois.

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois.

Date of amendment request: October 30, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to reduce the spent fuel pool (SFP) inadvertent draindown level to account for the effects of potential failures of the SFP cooling and skimmer loops.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

This change to the TS does not involve an increase in the probability of an accident previously evaluated. The initial conditions of the limiting dewatering incidents involve initiating circumstances/failures such as accidental gate openings, gate seal failures, or an open transfer tube.

Specifying a revised inadvertent drain limit which meets the SRP [Standard Review Plan, NUREG–0800] acceptance criteria is unrelated to the probability of occurrence of the precursors or initiating events. These initiators are not affected by the SFP cooling or skimmer loop piping/component failure scenarios. There is no change being made to the approved design, nor is there any operational change being made which would increase the probability of occurrence.

This change to the TS does not involve an increase in the consequences of an accident previously evaluated. As documented in NUREG–0876, Byron SER, Section 9.1.3, page 9–5, the anti-siphon protection design of the SFP cooling and clean-up piping was reviewed and found to be acceptable stating that "all connections to the spent-fuel pool are either near the normal water level or are provided with antisiphon holes to preclude possible siphon draining of the pool water." This review is applicable to Braidwood as documented in NUREG–1002, Braidwood SER. The anti-siphon attributes employed in the SFP skimmer loops at Braidwood, (under consideration at Byron), are similar in design as well as their submergence levels previously evaluated for the SFP cooling loops. The proposed change revises the SFP inadvertent drain limit from approximately 423 feet to 410 feet to bound the failure effects of both the SFP cooling and skimmer loops, while considering any maloperation or failure scenario. The revised value meets the SRP acceptance criteria of maintaining at least 10 feet above the active fuel ensuring that adequate radiation shielding is maintained as previously analyzed. There is no physical or operational change being made which would alter the sequence of events, plant response, or conclusions of the affected analysis. There is no change in the type or amount of any effluents released, and no change in either the Onsite or Offsite dose consequences as a result of this change.

Therefore, based on this evaluation, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change specifically identifies the SFP level sufficient to ensure that the SRP acceptance criteria for inadvertent draining are met while accounting for the failure effects of both the SFP cooling and skimmer loops. Any inadvertent SFP draining due to potential failures of the SFP skimmer loops is similar in nature to the inadvertent SFP draining effects previously considered due to failures of the SFP cooling loops. No new equipment is being installed, and no installed equipment is being operated in a new or different manner with this change. There is no change in plant operation that affects previously evaluated failure modes. This change does not represent a new failure mode or accident from what has been previously evaluated.

Therefore, the proposed change 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?

The current TS value does not address inadvertent SFP draining due to potential failures of the SFP skimmer loops or cooling suction lines as was done for the SFP cooling discharge lines. This change specifically identifies the SFP level sufficient to ensure that the SRP acceptance criteria for inadvertent draining are met while accounting for the failure effects of both the SFP cooling and skimmer loops in determining the proposed TS value. The most limiting postulated SFP dewatering incidents involve SFP drainage to either a dry transfer canal, a dry transfer canal and cask fill area, or a dry transfer canal and cask fill area which additionally communicates through an open transfer tube to an empty refuel cavity. The initial conditions of the dewatering incident analysis and resultant water levels over the spent fuel are not affected by this SFP skimmer/cooling loop issue because these incident initiators are not effected by the SFP cooling or skimmer loop failures, thus preserving the previously analyzed and approved margin for these dewatering incidents.

For the less-limiting SFP skimmer/cooling loop failure issue, the proposed TS change inadvertent drain limit meets the SRP minimum requirement of at least 10 feet above the top of the active fuel ensuring that adequate radiation shielding is maintained. This change would allow for the conservative acceptance criteria for the current UFSAR [Updated Final Safety Analysis Report] design analysis to continue to be met.

Therefore, this change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Stuart A. Richards.

Commonwealth Edison Company, Docket No. 50–374, LaSalle County Station, Unit 2, LaSalle County, Illinois.

Date of amendment request: November 9, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification 3/4.3.2, "Isolation Actuation Instrumentation" to add/revise various isolation setpoints for leak detection instrumentation. These changes are necessary due to modifications to the Reactor Water Cleanup (RWCU) System to restore "hot" suction to the RWCU pumps and due to a re-evaluation of the high energy line break analysis. In addition, the amendment would eliminate isolation actuation trip functions for the Residual Heat Removal (RHR) system steam condensing mode and shutdown cooling mode.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

(a) There is no effect on accident initiators so there is no change in probability of an accident. A line break in the subject areas, would consist of an instantaneous circumferential break downstream of the outermost isolation valve of one of these systems. The leak detection isolation is only a precursor of a break, and thus does not affect the probability of a break.

(b) There is minimal effect on the consequences of analyzed accidents due to changing the leak detection ambient temperature or Delta T setpoint and allowable values to detect 25 gpm equivalent leakage. The addition of more ambient temperature and ΔT leak detection monitoring, along with the addition of the high flow break detection will actually decrease the consequences of the associated accidents. The worst case accident outside the primary containment boundary is a main steam line break which bounds the dose consequences of all line breaks and therefore bounds any size of leak.

The deletion of the RHR steam condensing mode isolation actuation instrumentation trip functions from the LaSalle Technical Specifications does not increase the probability or consequences of an accident previously evaluated, because this mode of operation of the RHR system has been deleted from the LaSalle design basis and the lines that were previously high energy lines are isolated during unit operation, including Operational Condition 1 (Run mode), Operational Condition 2 (Startup mode), and Operational Condition 3 (Hot Shutdown).

The deletion of the RHR shutdown cooling mode leak detection T and Delta T isolation actuation instrumentation trip functions from the LaSalle Technical Specifications does not increase the probability or consequences of an accident previously evaluated, because the leak detection is only a precursor of a break, and thus does not affect the probability of a break. Also, there are two other methods of detecting abnormal leakage and isolating the system in Technical Specification trip functions A.6.a, Reactor Vessel Water Level—Low, Level 3 and A.6.c, RHR Pump Suction Flow—High. In addition, other means to detect leakage from the RHR system, such as sump monitoring and area radiation monitoring, are also available. In accordance with Technical Specification Administrative Requirement 6.2.F.1, LaSalle has a leakage reduction program to reduce leakage from those portions of systems outside primary containment that contain radioactive fluids. RHR, including piping and components associated with the shutdown cooling mode, is part of this program, which includes periodic visual inspection of the system for leakage. The sump monitoring, radiation monitoring and periodic inspections for system leakage makes the probability of a leak of 5 gpm going undetected for more than a day very low.

Also, due to the low reactor pressures (less than 135 psig) at which RHR shutdown cooling mode is able to operate, reactor coolant makeup and outflow is very low compared to normal plant operation. A change in flow balance due to a leak is thus more readily detectable with reactor coolant water level changes and makeup flow rate, and thus precludes a significant leak going undetected before break detection instrumentation would cause automatic isolation.

Therefore, this proposed amendment does not 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 because:

The purpose of the leak detection system, as it applies to the RWCU and RHR system areas, is to provide the capability for leak detection and automatic isolation of the system as necessary in the event of leakage in these areas. This change maintains this capability with at least two different methods of detection of abnormal leakage for protection from the flooding concerns of a significant leak or line break when the RHR system is operating in the shutdown cooling mode, so that redundant systems will not be affected.

This change also maintains or adds primary containment isolation logic for the leak detection isolation based on temperature monitoring in RWCU areas and break detection based on RWCU pump suction flow—high. The additional instrumentation and the associated isolation logic is the same or similar to existing instrumentation and logic for containment actuation instrumentation, so no new failure modes are created in this way.

Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Involve a significant reduction in the margin of safety because:

The change to the automatic isolation setpoint for high Delta T leak detection in the heat exchanger rooms is based on current configuration calculated/analyzed response to a small leak compared to a circumferential break. The increased leakage rate in the RWCU heat exchanger rooms that is necessary to actuate isolation on ambient temperature during winter conditions, does not adversely affect the margin of safety. This increased leakage rate is below the critical crack leakage rate as represented in UFSAR [Updated Final Safety Analysis Report] Figure 5.2–11. Additionally, differential temperature leak detection is conservative under these same conditions, and will actuate isolation at a leakage rate less than the established limit. The leak detection isolation logic is unchanged and thus remains single failure proof.

The addition of automatic primary containment isolation on ambient temperature and Delta T-High for the Reactor Water Cleanup System (RWCU) Pump, Pump Valve, Holdup Pipe, and Filter/Demineralizer (F/D) Valve Rooms and the addition of the RWCU Pump Suction Flow High line break isolation add to the margin of safety with respect to leak detection and line breaks in the RWCU system, because the system isolation diversity is increased and the amount of system piping monitored for leakage is increased.

The setpoints for the ambient temperature and Delta T leak detection isolations being changed or added and the RWCU pump suction flow—high are set sufficiently high enough so as not to increase the possibility of spurious actuation. In the event that a spurious actuation does occur, little safety significance is presented since the RWCU system performs no safety function. The setpoints and allowable values for the proposed changes also assure sufficient margin to the analytical values and are high enough to prevent spurious actuations based on calculations consistent with Regulatory Guide 1.105.

The deletion of the RHR steam condensing mode isolation actuation instrumentation does not effect the margin of safety, because this mode is no longer utilized by LaSalle in Operational Conditions 1, 2, or 3 (Run mode, Startup mode, or Hot Shutdown).

The elimination of the temperature based trip functions for the RHR shutdown cooling mode area is based on the determination that temperature is not the appropriate parameter for leak detection as it does not provide meaningful indication and will not provide setpoints that would be sufficiently above the normal range of ambient conditions to avoid spurious isolations.

There are two other methods of detecting abnormal leakage and isolating the system in Technical Specification trip function A.6, which are A.6.a, Reactor Vessel Water Level—Low, Level 3 and A.6.c, RHR Pump Suction Flow—High. In addition, other means to detect leakage from the RHR system, such as sump monitoring and area radiation monitoring, are also available. Also, in accordance with Technical Specification Administrative Requirement 6.2.F.1, LaSalle has a leakage reduction program to reduce leakage from those portions of systems outside primary containment that contain radioactive fluids. RHR, including piping and components associated with the shutdown cooling mode, is part of this program, which includes periodic visual inspection of the system for leakage.

The previous evaluation of diversity of isolation parameters, as presented in Table 5.2–8 of the UFSAR remains unchanged. Adequate diversity of isolation parameters is maintained because there are at least two different methods available to detect and allow isolation of the system for a line break, as necessary.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendment involves no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Stuart A. Richards.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York.

Date of amendment request: October 9, 1998.

Description of amendment request: The proposed amendment would revise Section 6.0, administrative controls, of the Technical Specifications (TSs). Specifically, TS Sections 6.5.2.1.j, 6.7.1.c, and 6.8.1.a would be revised to correct typographical errors. In addition, TS Section 6.5.2.2 would be revised to change the membership of the Nuclear Facility Safety Committee (NFSC). This change would provide Consolidated Edison (Con Ed) with the flexibility to obtain industry experts outside of Con Ed to perform the duties of Chairman, or Vice Chairman, and members of the NFSC.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. There is no significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment is administrative in nature. It involves a change in 1) the Nuclear Facilities Safety Committee (NFSC) Chairman or Vice Chairman to allow the services of an individual other than a senior official of the Company, and 2) allowing NFSC membership by other than Con Edison employees. In either case, concurrence by the Senior Vice President, Nuclear Operations is required.

These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operating of the facility. The Limiting Safety Systems Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes to the subject Technical Specification would not increase the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

As stated above, the proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility, and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently, no new failure modes are introduced as a result of the proposed changes. Therefore, the proposed changes will not initiate any new or different kind of accident.

3. There has been no significant reduction in the margin of safety.

The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or physical design the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed changes will not result in a reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: S. Singh Bajwa, Director.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan.

Date of amendment request: November 9, 1998.

Description of amendment request: The proposed amendment would delete the Chemical and Volume Control System (CVCS) operability requirements currently in technical specifications (TS) 3.2 and 3.17.6, and the associated surveillance testing requirements currently in TS 4.2 and 4.17. The requirements have been added to the Palisades Operating Requirements Manual (ORM).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes delete certain TS requirements which do not meet the criteria of 10 CFR 50.36(c)(2)(ii), but identical requirements have been added to a document (the ORM) controlled under 10 CFR 50.59.

10 CFR 50.59 specifically prohibits changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report "if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased". Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, moving of a requirement from the TS to a document which is controlled under 50.59 cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?

The proposed changes delete certain TS requirements which do not meet the criteria of 10 CFR 50.36(c)(2)(ii), but identical requirements have been added to a document (the ORM) controlled under 10 CFR 50.59.

10 CFR 50.59 specifically prohibits changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report "if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created". Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to a document which is controlled under 50.59 cannot create the possibility of a new or different kind of accident from any previously evaluated.

Do the proposed changes involve a significant reduction in a margin of safety?

The proposed changes delete certain TS requirements which do not meet the criteria of 10 CFR 50.36(c)(2)(ii), but identical requirements have been added to a document (the ORM) controlled under 10 CFR 50.59.

10 CFR 50.59 specifically prohibits changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report if the margin of safety is reduced. Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to a document which is controlled under 50.59 cannot involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423–3698.

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: Cynthia A. Carpenter.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina.

Date of amendment request: July 22 and October 22, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to reflect the licensee's planned use of fuel supplied by Westinghouse. The Westinghouse fuel has different design characteristics from the fuel currently in use. Accordingly, the following changes would need to be made to the TS: Figure 2.1.1–1, ''Reactor Core Safety Limits—Four Loops in Operation''; various core operating parameters specified by Surveillance Requirements 3.2.1.2, 3.2.1.3, and 3.2.2.2; Section 4.2.1, ''Fuel Assemblies''; and Section 5.6.5, ''Core Operating Limits Report (COLR).''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

First Standard

Implementation of this LAR [license amendment request] would not involve a significant increase in the probability or consequences of an accident previously evaluated. The revised Reactor Core Safety Limits Figure further restricts acceptable operation. Moving an uncertainty factor from the Improved Technical Specifications to the Core Operating Limits Report (COLR) does not exempt this factor from regulatory restrictions. COLR parameters are generated by NRC approved methods with the intent of ensuring that previously evaluated accidents remain bounding. The COLR is submitted to the NRC upon implementation of each fuel cycle or when the document is otherwise revised. No accident probabilities or consequences will be impacted by this LAR.

Second Standard

Implementation of this LAR would not create the possibility of a new or different kind of accident from any previously evaluated. The revised Reactor Core Safety Limits Figure further restricts acceptable operation. Moving an uncertainty factor from the Improved Technical Specifications to the COLR does not exempt this factor from regulatory restrictions. Since the parameter in question is not being deleted, the possibility of a new or different kind of accident from any previously evaluated does not exist.

Third Standard

Implementation of this LAR would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. Use of the ZIRLO™ cladding material has been reviewed and approved in Reference 1 (as listed in Chapter 2.1 of Topical Report DPC–NE–2009/DPC–NE–2009P, Duke Power Company Westinghouse Fuel Transition Report). ZIRLO™ cladding has been extensively used in Westinghouse nuclear reactors. The changes proposed in this LAR are necessary to ensure that the performance of the fission product barriers (cladding) will not be impacted following the replacement of one fuel design for another. No safety margin will be significantly impacted.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana.

Date of amendment request: November 20, 1998.

Description of amendment request: The licensee has proposed an amendment to Facility Operating License No. NPF–47, Appendix A—Technical Specifications (TS) Section 3.1.6, ''Control Rod Pattern.'' The proposed change will be implemented through the establishment of a new specification added to Section 3.10, ''Special Operations.'' The proposed specification will be TS Section 3.10.9, ''Control Rod Pattern—Cycle 8.'' The new TS 3.10.9 is required due to a current plant-specific configuration where 5 control rods have been inserted into the reactor core for neutron flux suppression surrounding 2 fuel assemblies which have been identified as having possible fuel cladding defects. The new requirement is intended to be effective for the remainder of the current fuel cycle (Cycle 8), and is in force when rod withdrawal operations begin from a condition of 100% rod density to 20% rated thermal power (RTP).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) The request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Accidents analyzed in the SAR have been examined for any impact caused by this exception to the [Banked Position Withdrawal Sequence] BPWS operation. The limiting event is the [Control Rod Drive Accident] CRDA as described in SAR Sections 4.3.2 and 15.4.9. The limit on energy addition to the fuel is 280 cal/gm as identified in the SRP section 15.4.9. Bank Position Withdrawal Sequence is established to reduce maximum incremental control rod worths and thus minimize consequences resulting from an accident. The reactor will be operated as before using BPWS. Having the current rod configuration with 5 rods to minimize impact on the two fuel cladding imperfections, in lieu of eight rods inoperable separated by two cells, will not affect initiators of a Control Rod Drop Accident. In addition, this existing rod configuration has been analyzed and the resulting consequences continue to be bounded by the licensing evaluations. The insertion of the identified control rods will not affect the assumed reactivity insertion time of any event. The location of the control rods has been reviewed by GE using the NRC approved methodology. Operation within these limits will ensure that the consequences of a transient or accident remain within the acceptable limits of the evaluation. Specifically, rod worths for the proposed configuration are bounded by the rod worths allowed for these configurations per TS; thus, the proposed configuration is more conservative than that allowed per TS. The results confirm all assumed limits are maintained. The proposed change ensures that the consequences of abnormal operation and accidents are acceptable.

The additional Technical Specification will control the configuration of the plant to that supported by the evaluation. If this evaluated configuration is not supported, the plant will be required to be placed in a configuration where the Control Rod Drop Accident is not applicable, as the current specification requires. The plant is therefore maintained within limits as currently allowed. With these limits the consequences of an event are not increased.

The probability of an accident is not affected by the proposed Technical Specification changes since the operation of systems or equipment that could initiate an accident are not affected. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

(2) The request does not create the possibility of occurrence of a new or different

kind of accident from any accident previously evaluated.

The proposed changes do not involve any alteration of plant hardware or significant change in plant operation. Assuming the 5 suppression rods are bypassed in lieu of eight rods separated by two cells does not affect event initiators or event consequences. No plant modifications are required which would affect plant operation. Operation with the control rod pattern in the proposed configuration will ensure the results of a CRDA will remain within the assumptions of the current safety analysis. The system will continue to ensure that the limits of control rod worth remain within the assumptions of the CRDA. The revised Technical Specifications will continue to assure that plant operation is consistent with the assumptions, initial conditions, and assumed power distribution and, therefore, will not create a new type of accident.

The proposed Technical Specifications will maintain the plant in a configuration supported by evaluation. The response to a CRDA will be within current accepted limits and therefore no event of a different kind has been created. The proposed Technical Specification changes do not introduce any new modes of plant operation nor involve new system interactions. Therefore, operation with the 5 suppression rods inserted does not create the possibility of an occurrence of a new or different kind of accident from any accident previously evaluated.

(3) The request does not involve a significant reduction in a margin of safety.

The proposed Technical Specification and the rod pattern control system will continue to ensure the limits of control rod worth remain within the assumptions which support the CRDA analysis of 280 cal/gm maximum energy heat addition to the fuel. This imposed limit of 280 cal/gm provides a margin of safety from the experimental value of approximately 330 cal/gm at which the fully molten state for UO_2 occurs. The existing rod configuration with 5 suppression rods inserted to minimize impact on the two fuel cladding imperfections has been analyzed using NRC approved methodology. Cycle specific evaluation has confirmed that the consequences resulting from a CRDA continues to be bounded by the licensing analysis for this event. Since there are no changes in the acceptance criteria, the proposed changes will not create a reduction in the margin of safety. These limits establish the necessary restrictions on power operation and thereby ensure that the core is operated within the assumptions and initial conditions of the transient and accident analyses.

As demonstrated in the evaluation, operation within these limits will ensure that the margin of safety will be maintained to the same level described in the Technical Specifications Bases and the USAR and the consequences of the postulated transient or accidents are not increased. This limit of 280 cal/gm is not exceeded during any transient or postulated accident. Therefore, the proposed Technical Specifications to allow startup and continued operation in the low power region with these control rods inserted do not involve a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Project Director: John N. Hannon.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana.

Date of amendment request: July 2, 1998.

Description of amendment request: The proposed change will modify the ACTION Requirements for Technical Specification (TS) 3/4.3.2 for the Emergency Feedwater Actuation Signal (EFAS). A change to the TS Bases Section 3/4.3.2 has been included to support this change. The objective of this change is to add a restriction on the period of time a channel of EFAS instrumentation can remain in the tripped condition.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed revision to the TS changes the allowed outage time that a channel of EFAS SGDPI [Steam Generator Differential Pressure Instrumentation] can be in the tripped condition from a maximum of approximately 18 months when one channel is inoperable and 92 days when two channels are inoperable to 48 hours. If a channel were in the tripped condition and a single failure occurred (failure of one other channel of EFAS SGDPI), an inadvertent EFAS signal would be generated. During a Design Basis MSLB [Main Steam Line Break] or FLB [Feedwater Line Break] Accident, this single failure would send EFW [Emergency Feedwater] to the faulted steam generator. The Waterford 3 safety analysis assumes that the excess Reactor Coolant System (RCS) cooldown and return to power associated with the MSLB will be terminated when the faulted steam generator empties. If additional EFW were added, the RCS cooldown would be extended and the return to power may increase.

Reducing the time that a channel of EFAS SGDPI can be placed in the tripped condition will reduce the probability of this scenario occurring during a Design Basis Accident. Since the allowed outage time for a channel of EFAS SGDPI is being limited to 48 hours, this is considered an off-normal operation and a single failure is not required to be postulated during a Design Basis Accident in the accident analysis. Reducing the time the channel can be placed in the tripped condition and thus, the exposure time to this scenario, would not be an accident initiator. The proposed change of being more conservative relative to allow[ed] outage time in the tripped condition will not affect the assumptions, design parameters, or results of any accident previously evaluated.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

The proposed change does not alter the design or configuration of the plant. The proposed change provides a more conservative allowed outage time for the channel to be in the tripped condition. There has been no physical change to plant systems, structures or components nor will the proposed change reduce the ability of any of the safety-related equipment required to mitigate Anticipated Operational Occurrences or accidents. The configuration required by the proposed specification is permitted by the existing specification.

Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change provides a more conservative allowed outage time for the channel to be in the tripped condition. By reducing the allowed outage time, the probability is reduced that a single failure (failure of one channel of EFAS SGDPI with one channel in the tripped condition) would occur that would send EFW to the faulted steam generator. Therefore, the only change to the margin of safety would be an increase. Since the allowed outage time for a channel of EFAS SGDPI is being limited to 48 hours, this is considered an off-normal operation and a single failure is not required to be postulated during a Design Basis Accident in the accident analysis. The proposed changes do not affect the limiting conditions for operation or their bases.

Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Project Director: John N. Hannon.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida.

Date of amendment request: December 31, 1997, as supplemented November 25, 1998.

Description of amendment request: The proposed amendment will revise the St. Lucie Unit 2 Technical Specifications to permit an increase in the allowed Spent Fuel Pool (SFP) storage capacity.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Analyses to support the proposed fuel pool capacity increase have been developed using conservative methodology. The analysis of the potential accidents summarized below has shown that there is no significant increase in the consequences of any accident previously analyzed. A review of relevant plant operations has also demonstrated that there is no significant increase in the probability of occurrence of any accident previously analyzed. This conclusion is also discussed below.

Previously evaluated accidents that were examined for this proposed license amendment include: Fuel Handling Accident, Spent Fuel Cask Drop Accident, and Loss of all Fuel Pool Cooling.

There will be no change in the mode of plant operation or in the availability of plant systems as a result of this proposed change; the systems interfacing with the spent fuel pool have previously encountered borated pool water and are designed to interact with irradiated spent fuel and remove the residual heat load generated by isotopic decay. The proposed amendment does not require a change in the maintenance interval or maintenance scope for the fuel pool cooling system or for the spent fuel cask crane. The frequency of cask handling operations and the maximum weight carried by the crane is not increased as a result of the proposed license amendment. Thus, there will be no increase in the probability of a loss of fuel pool cooling or in the probability of a failure of the cask crane as a result of the proposed amendment.

There will not be a significant increase in the frequency of handling discharged assemblies in the fuel pool as a result of this change; any handling of fuel in the spent fuel pool will continue to be performed in borated water. If the license amendment is approved, there will be a one-time repositioning of certain discharged assemblies stored in the fuel pool to comply with the revised positioning requirements, but the increased pool storage capacity will permit the deferral of spent fuel handling associated with cask loading operations. Fuel manipulation during the repositioning activity will be performed in the same manner as for fuel placed in the spent fuel pool during refueling outages. There will be no changes in the manner of handling fuel discharged from the core as a result of refueling; administrative controls will continue to be used to specify fuel assembly placement requirements. The relative positions of Region I and Region II storage locations will remain the same within the fuel pool. Therefore, the probability of a fuel handling accident has not been significantly increased.

The consequences of a fuel handling accident have been evaluated. The radioactive release consequences of a dropped fuel assembly are not affected by the proposed increase in fuel pool storage capacity. They remain bounded by the results of calculations performed to justify the existing St. Lucie Unit 2 fuel storage racks and burnup limits. At the limiting fuel assembly burnup, radioactive releases from a dropped assembly would be only a small fraction of NRC guidelines. The input parameters employed in analyzing this event are consistent with the current values of fuel enrichment, discharge burnup and uranium content used at St. Lucie Unit 2 and with future use of the ''value-added'' fuel pellet design. Thus, the consequences of the fuel assembly drop accident would not be significantly increased from those previously evaluated.

The capability of the fuel pool cooling system to handle the increased number of discharged assemblies has been examined. The impact of a total loss of spent fuel pool cooling flow on available equipment recovery time and on fuel cladding integrity has also been evaluated. For the limiting full core discharge, sufficient time remains available to restore cooling flow or to provide an alternate makeup source before boiloff results in a fuel pool water level less than that needed to maintain acceptable radiation dose levels. Analysis has shown that in the event of a total loss of fuel pool cooling fuel cladding integrity is maintained. Therefore, the consequences of a loss of fuel pool cooling event, including the effect of the proposed increase in fuel pool storage capacity, have not been significantly increased from previously analyzed results for this type of accident.

The analysis of record pertaining to the radiological consequences of the hypothetical drop of a loaded spent fuel cask just outside the Fuel Handling Building was examined to determine the impact of the increased fuel storage capacity on this accident's results. The results of the previously performed analysis were determined to bound the conditions described by the proposed license amendment, thus the consequences of the cask drop accident would not be significantly increased as a result of this change.

It is concluded that the proposed amendment to increase the storage capacity of the St. Lucie Unit 2 spent fuel pool will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different type of accident from any accident previously evaluated.

In this license amendment FPL [Florida Power & Light Co.] proposes to credit the negative reactivity associated with a portion of the soluble boron present in the spent fuel pool. Soluble boron has always been present in the St. Lucie Unit 2 spent fuel pool; as such the possibility of an inadvertent fuel pool dilution has always existed. However, the spent fuel pool dilution analysis demonstrates that a dilution of the Unit 2 spent fuel pool which could increase the pool k_{eff} to greater than 0.95 is not a credible event. Neither implementation of credit for the reactivity of fuel pool soluble boron nor the proposed increase in the fuel pool storage capacity will create the possibility of a new or different type of accident at St. Lucie Unit 2.

An examination of the limiting fuel assembly misload has determined that this would not represent a new or different type of accident. None of the other accidents examined as a part of this license submittal represent a new or different type of accident; each of these situations has been previously analyzed and determined to produce acceptable results.

The proposed license amendment will not result in any other changes in the mode of spent fuel pool operation at St. Lucie Unit 2 or in the method of handling irradiated nuclear fuel. The spatial relationship between the fuel storage racks and the cask crane range of motion is not affected by the proposed change.

As a result of the evaluation and supporting analyses, FPL has determined that the proposed fuel pool capacity increase does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

FPL has determined, based on the nature of the proposed license amendment that the issue of margin of safety, when applied to this fuel pool capacity increase, should address the following areas:

1. Fuel Pool reactivity considerations
2. Fuel Pool boron dilution considerations
3. Thermal-Hydraulic considerations
4. Structural loading and seismic considerations

The Technical Specification changes proposed by this license amendment, the proposed spent fuel pool storage configuration and the existing Technical Specification limits on fuel pool soluble boron concentration provide sufficient safety margin to ensure that the array of fuel assemblies stored in the spent fuel pool will always remain subcritical. The revised spent fuel storage configuration is based on a Unit 2 specific criticality analysis performed using methodology consistent with that approved by the NRC. Additionally, the soluble boron concentration required by current Technical Specifications ensures that the fuel pool k_{eff} will be always be maintained substantially less than 0.95.

The Unit 2 criticality analysis established that the k_{eff} of the spent fuel pool storage racks will be less than 1.0 with no soluble boron in the fuel pool water, including the effect of all uncertainties and tolerances. Credit for the soluble boron actually present is used to offset uncertainties, tolerances, off-normal conditions and to provide margin such that the spent fuel pool k_{eff} is maintained less than or equal to 0.95. FPL has also demonstrated that a decrease in the fuel pool boron concentration such that k_{eff} exceeds 0.95 is not a credible event.

Current Technical Specifications require that the fuel pool boron concentration be maintained greater than or equal to 1720 ppm. This boron value is substantially in excess of the 520 ppm required by the uncertainty and reactivity equivalencing analyses discussed in this evaluation and the 1266 ppm value required to maintain k_{eff} less than or equal to 0.95 in the presence of the most adverse mispositioned fuel assembly.

The St. Lucie Unit 2 fuel pool boron concentration will continue to be maintained significantly in excess of 1266 ppm; the proposed license amendment will not result in changes in the mode of operation of the refueling water tank (RWT) or in its use for makeup to the fuel pool. Thus, operation of the spent fuel pool following the proposed change, combined with the existing fuel pool boron concentration Technical Specification limit of 1720 ppm, will continue to ensure that k_{eff} of the fuel pool will be substantially less than 0.95.

Even if this not-credible dilution event was to occur, no radiation would be released; the only consequence would be a reduction of shutdown margin in the fuel pool. The volume of unborated water required to dilute the fuel pool to a k_{eff} of 0.95 is so large (in excess of 358,900 gallons to dilute the fuel pool to 520 ppm boron) that only a limited number of water sources could be considered potential dilution sources. The likelihood that this level of water use could remain undetected by plant personnel is extremely remote.

In meeting the acceptance criteria for fuel pool reactivity, the proposed amendment to increase the storage capacity of the existing fuel pool racks does not involve a significant reduction in the margin of safety for nuclear criticality.

Calculations of the spent fuel pool heat load with an increased fuel pool inventory were performed using ANSI/ANS–5.1–1979 methodology. This method was demonstrated to produce conservative results through benchmarking to actual St. Lucie Unit 2 fuel pool conditions and by comparison of its results to those generated by a calculation using Auxiliary Systems Branch Technical Position 9–2 methodology. Conservative methods were also used to demonstrate fuel cladding integrity is maintained in the absence of cooling system forced flow. The results of these calculations demonstrate that, for the limiting case, the existing fuel pool cooling system can maintain fuel pool conditions within acceptable limits with the increased inventory of discharged assemblies.

Therefore, the proposed change does not result in a significant reduction in the margin of safety with respect to thermal-hydraulic or spent fuel cooling considerations.

The primary safety function of the spent fuel pool and the fuel storage racks is to maintain discharged fuel assemblies in a safe configuration for all environments and abnormal loadings, such as an earthquake, a loss of pool cooling or a drop of a spent fuel assembly during routine spent fuel handling. The proposed increase in spent fuel inventory on the fuel pool and the existing storage racks have been evaluated and show that relevant criteria for fuel rack stresses and floor loadings have been met and that there has been no significant reduction in the margin of safety for these criteria.

The NRC staff has reviewed the licensee's analysis and the changes proposed in the November 25, 1998 supplement to the original submittal and based on this review, it appears that the three standards of 50.92(c) continue to be satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Project Director: Frederick J. Hebdon.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida.

Date of amendment request: October 27, 1998.

Description of amendment request: The licensee proposed to change Technical Specification (TS) 6.3, Facility Staff Qualifications, in order to incorporate qualifications for the Multi-Discipline Supervisor. The current TS requires that plant staff meet the requirements of the American National Standards Institute (ANSI) N18.1–1971, which requires non-licensed supervisors to have a high school diploma or equivalent and a minimum of 4 years experience in the craft or discipline they supervise. The proposed change requires the Multi-Discipline Supervisor to have, (1) a high school diploma or equivalent, (2) a minimum of 4 years of related technical experience, which shall include 3 years of power plant experience of which one year is at a nuclear power plant, and (3) completed the Multi-Discipline Supervisor training program.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature addressing personnel qualification issues. The Multi-Discipline Supervisor (MDS) position will be filled with personnel who are experienced in one or more technical disciplines (maintenance, operations, engineering, or other related technical discipline). Fundamental working knowledge of tasks being performed will be acquired through the MDS initial training program. The training concentrates on developing the skills and knowledge of an MDS to safely oversee tasks for multi-discipline work teams. Therefore, four years experience in any related technical discipline or disciplines combined with the MDS training program provide adequate technical knowledge for proper job oversight. These proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes being proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or modes of plant operation defined in the facility operating license, or Technical Specifications that preserve safety analysis assumptions. These changes address qualification requirements for the MDS position. Since the proposed changes do not change the qualifications for those individuals responsible for the actual licensed operation of the facility, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new failure mode is introduced due to the administrative changes since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems, structures, or components.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The proposed changes to add the MDS position have management and administrative controls associated with the required qualification requirements. The Turkey Point Technical Specifications will ensure that any individual filling the MDS position has the requisite education, experience, and training. As a result, operation of the facility in accordance with the proposed changes would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Project Director: Frederick J. Hebdon.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey.

Date of amendment request: November 5, 1998.

Description of amendment request: The proposed Technical Specification change will modify the safety limits and surveillances of the LPRM and APRM systems and related Bases pages to ensure the APRM channels respond within the necessary range and accuracy and to verify channel operability. In addition, an unrelated change to the Bases of Specification 2.3 is included to clarify some ambiguous language.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed technical specification changes to the limits and surveillance requirements of the LPRM and APRM systems are provided to ensure the APRM channels respond within the necessary range and accuracy and to verify channel operability. If one or more monitored parameters exceeded their specified limits, the RPS initiates a reactor scram signal to preserve the integrity of the fuel cladding and the Reactor Coolant System and minimize the energy that must be absorbed following a loss of coolant accident. Therefore, the probability of occurrence or the consequences of an accident previously evaluated in the [safety analysis report] SAR will not increase as a result of these changes.

2. The proposed technical specification changes to the limits and surveillance requirements of the LPRM and APRM systems are provided to ensure the APRM channels respond within the necessary range and accuracy and to verify channel operability. The proposed changes are designed to ensure the APRM system responds in a manner that ensures the safety limits, limiting safety system settings, limiting conditions for operations, as well as design parameters for the APRM system and individual components are continuously met. Therefore, the proposed activity does not create the possibility for an accident or malfunction of a different type than any previously identified in the SAR.

3. The proposed change does not involve a significant reduction in the margin of safety. When the APRMs exceed their specified limits, the RPS initiates a reactor scram signal to preserve the integrity of the fuel cladding and the Reactor Coolant System and minimize the energy that must be absorbed following a loss of coolant accident. The proposed changes are designed to assure the APRM system responds in a manner that ensures the safety limits, limiting safety system settings, limiting conditions for operations, as well as design parameters for the APRM system and individual components are continuously met. Therefore, the margin of safety will not be reduced.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cecil O. Thomas.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania.

Date of amendment request: November 25, 1998.

Description of amendment request: The proposed amendment will change the surveillance specification for Once Through Steam Generator (OTSG) inservice inspections for TMI–1 Cycle 13 refueling outage examinations which would be applicable for the next operating cycle only, Operating Cycle 13.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

A. The proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.

The proposed flaw disposition strategy, based on measurable eddy current parameters of axial and circumferential extent for Inside Diameter (ID) Initiated Inter-Granular Attack (IGA), will continue to provide high confidence that unacceptable flaws that do not have the required structural integrity to withstand a postulated MSLB [main steam line break] are removed from service. The axial and circumferential length limits for eddy current ID degradation indications meet the Draft Regulatory Guide 1. 121 * * * acceptance criteria for margin to failure for MSLB-applied differential pressure and axial tube loads. The capability for detection of flaws is unaffected; and the identification of tubes that should be repaired or removed from service is maintained. The operation of the OTSGs or related structures, systems, or components is otherwise unaffected. Therefore, neither the probability nor consequences of [an] SGTR [steam generator tube rupture] is significantly increased either during normal operation or due to the limiting loads of [an] MSLB accident.

Neither the change in voltage normalization for the eddy current examinations, nor the administrative change in clarification of the reporting requirements, as described above, could significantly affect the probability of occurrence or consequences of any accident previously evaluated. These changes are administrative only.

B. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no hardware changes involved nor changes to any operating practices. These changes involve only the OTSG tube inservice inspection surveillance requirements, which could only affect the potential for OTSG primary-to-secondary leakage. The proposed changes continue to impose flaw length limits for ID IGA to assure tube structural and leakage integrity, as confirmed by 12R (and post 12R) tube pull sample examinations and pressure testing.

In addition, neither the change in voltage normalization for the eddy current examinations nor the administrative change in the description of the reporting requirements, as described above, could possibly create the possibility of an accident of a new or different type from any previously evaluated. These changes are included only to modify the plant's eddy current normalization to the industry standard, and clarify the reporting period for submittal of the OTSG inspection results to the NRC [Nuclear Regulatory Commission]. Therefore, these changes do not create the potential for any other kind of accident different from those that have been evaluated.

C. These proposed changes do not involve a significant reduction in a margin of safety because the margins of safety defined in Draft Regulatory Guide 1. 121 * * * are retained. The probability of detecting degradation is unchanged since the bobbin coil eddy current methods will continue to be the primary means of initial detection and the probability of leakage from any indications left in service remains acceptably small. The strategy for dispositioning ID initiated IGA will continue to provide a high level of confidence that tubes exceeding the allowable limits for tube integrity are repaired or removed from service.

In addition, neither the change in voltage normalization for the eddy current examinations nor the administrative change in the description of the reporting requirements, as described above, could significantly affect a margin of safety. These changes are administrative in nature and are included only to align TMI–1's voltage normalization to the industry standard, and clarify the reporting period, respectively.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (Regional Depository) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cecil O. Thomas.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut.

Date of amendment request: November 10, 1998.

Description of amendment request: The proposed changes would modify Technical Specifications 3.3.1.1, ''Reactor Protective Instrumentation,'' and 3.3.2.1, ''Engineered Safety Feature Actuation System Instrumentation'' to restrict the time a reactor protection or engineered safety feature actuation channel can be in the bypass position to 48 hours, from an indefinite period of time. Most of these proposed changes were originally submitted in a letter dated May 14, 1998. The licensee withdrew its original request and submitted a new request in its November 10, 1998, letter.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

In accordance with 10CFR50.92, NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes and has concluded that they do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to restrict the time [* * *] reactor protection or engineered safety feature actuation channels can be in the bypass position to 48 hours, from an indefinite period of time, has no effect on the design of the Reactor Protection System (RPS) or the Engineered Safety Feature Actuation System (ESFAS) and does not affect how these systems operate. In addition, this will minimize the susceptibility of these systems to the remote possibility of fault propagation between channels. However, this proposed change will require an inoperable pressurizer high pressure reactor protection channel to be placed in the tripped condition within 48 hours. With a pressurizer pressure channel in the tripped condition, the high failure of a second pressurizer pressure channel would initiate a reactor trip and open both pressurizer power operated relief valves (PORVs). Opening the pressurizer PORVs would result in an undesired loss of primary coolant. Thus, this change will increase the probability of occurrence of a previously evaluated accident. However, this would not place the plant in an unanalyzed condition since FSAR [Final Safety Analysis Report] Section 14.6.1 analyzes the inadvertent opening of both PORVs, the release of reactor coolant can be terminated by closure of the PORV block valves from the control room, and the Emergency Operating Procedures provide guidance on how to address this situation. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to increase the time a second RPS or ESFAS channel can be removed from service (from 2 hours to 48 hours), provided one of the inoperable channels is placed in the tripped condition, has no effect on the design of the RPS or ESFAS and does not affect how these systems operate. These systems will still function as designed to mitigate design basis accidents. However, this change will also impact the probability of occurrence of a previously evaluated accident since it will allow a second pressurizer high pressure reactor protection channel to be placed in the tripped condition for 48 hours instead of the current 2 hour time limit. The impact of this change is bounded by the proposed change to require an inoperable pressurizer high pressure reactor protection channel to be placed in the tripped condition after 48 hours as previously discussed. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to apply a more restrictive action statement to the loss of turbine load reactor trip function has no effect on the design of this trip function and does not affect how this trip function operates. Also, this trip function is not assumed to operate to mitigate any design basis accident. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to require a channel calibration every 18 months for the loss of turbine load reactor trip function and for the wide range logarithmic neutron flux monitors has no effect on the design of either the loss of turbine load reactor trip function or the wide range logarithmic neutron flux monitors. Also, neither of these are assumed to operate to mitigate any design basis accident. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to exclude the neutron detectors from the channel calibration requirement has no effect on the design of the neutron detectors and has no significant effect on how these detectors operate. The detectors are passive devices with minimal drift. In addition, slow changes in the sensitivity of the linear power range flux detectors is compensated for by performing the daily calorimetric calibration and the monthly calibration using the incore detectors. These detectors will still function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to add the license amendment numbers to Technical Specification Page 3/4 3–9 will not result in a technical change to the Millstone Unit No. 2 Technical Specifications. The RPS will continue to function as before. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to correct the surveillance requirement referenced in an action statement has no effect on the design of the ESFAS and does not affect how this system operates. The ESFAS will still function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to add a reference to the reactor coolant pump low speed reactor trip function to a note that states this trip may be bypassed <5% power, and that the bypass must be automatically removed [greater than or equal to] 5% will not affect this reactor trip function. This bypass capability currently exists in the design of the Millstone Unit No. 2 RPS, and is the same bypass feature referenced for the reactor coolant flow low reactor trip function. Both of these reactor trip functions provide protection for a reduction in RCS [reactor coolant system] flow. The addition of this note will not result in any technical change to the Millstone Unit No. 2 RPS. The RPS will continue to function as before. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to correct the power level high trip setpoint on Technical Specification Page 2–4 will not result in any change to the actual plant setpoint for this RPS trip function. As a result of this proposed change, the setpoint listed on Page 2–4 will agree with the setpoint previously approved by the NRC, and currently used by the RPS. The change has no effect on the design of the RPS and does not affect how this system operates. The RPS will still function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The information added to the Bases of the affected Technical Specifications to provide a discussion of how the RPS and ESFAS are affected by the proposed changes, the effect the action statements have on the operation of the RPS and ESFAS, and to discuss the impact of surveillance testing on RPS operability will have no effect on equipment operation. The RPS and ESFAS will continue to function as designed to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

Thus, this License Amendment Request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. They will not alter assumptions made in the safety analysis and licensing basis. The RPS and the ESFAS will still function as designed to mitigate design basis accidents.

Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes will not reduce the margin of safety since they have no impact on any safety analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor do the proposed changes affect any instrument setpoints or equipment safety functions.

The effectiveness of Technical Specifications will be maintained since the changes will not alter the operation of any RPS or ESFAS function. In addition, most of the changes are consistent with the Calvert Cliffs RPS and ESFAS Technical Specifications model provided in Enclosure 3 of the NRC correspondence dated April 16, 1981 [R. A. Clark letter to W. G. Council, Evaluation of the Reactor Protection System Inoperable Channel Condition at Millstone Nuclear Power Station, Unit No. 2, dated April 16, 1981] and with the new, improved Standard Technical Specifications (STS) for Combustion Engineering plants (NUREG–1432).

Therefore, there is no significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Project Director: William M. Dean.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota.

Date of amendment request: November 25, 1997, as supplemented September 25 and November 11, 1998. The September 25, 1998, supplement incorrectly references the original request as October 31, 1997, rather than November 25, 1997.

Description of amendment request: The proposed amendment would revise the Technical Specifications for the condensate storage tank (CST) low level suction transfer setpoint for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems to allow removing one CST from service for maintenance.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed setpoint change and temporary level switch cross connection will not affect the way the suction transfer equipment functions, introduce new failure modes, or significantly increase the probability of failure of this equipment.

A slight increase in the probability of failure of the CST suction low level automatic transfer function may result, however, during plant operation with one CST in service and the CST low level transfer switches temporarily cross connected. This temporary modification preserves the redundancy of the automatic level transfer logic and allows HPCI and RCIC to remain aligned to the condensate storage system.

When the switches are cross connected, sections of piping and instrument tubing will be shared by both level switches. The probability that freezing or plugging of a common section of piping or tubing will disable both switches will be slightly higher than during two CST operation with the level switch piping in its normal configuration.

The level switches would be cross connected at infrequent intervals to permit prudent and timely CST preventive maintenance and at the same time continue to provide HPCI and RCIC with a source of reactor makeup quality water. In the unlikely event of a spurious actuation of either system, only high quality water would be injected into the reactor vessel.

Overall, the possibility of freezing or plugging of piping and tubing associated with the automatic transfer level switches has been shown to be very small, with or without the temporary level switch cross connection in place. During periods of operation with one CST, we believe the small additional opportunity for level instrument failure due to freezing or plugging is more than compensated for by the benefits of maintaining a high quality source of water to the HPCI and RCIC pumps.

The proposed level switch cross connection will not affect the way the suction transfer equipment functions. The cross connection tubing will be evaluated for seismic loads equivalent to the existing instrument piping. Rupture of the tubing will not prevent the function of the level switches from being accomplished and no other equipment important to safety is impacted by these changes.

Technical Specification and other specified margins of safety are effectively increased by the proposed changes. The HPCI/RCIC low CST level suction transfer level is being adjusted upward in the conservative direction.

The changes do not present the opportunity for a new release path for radioactive material.

These changes have no impact on the protection of the health and safety of the public.

(2) The proposed amendment will not create the possibility of a new or different

kind of accident from any accident previously analyzed.

No system, structure, or component (SSC) described in the USAR [Updated Safety Analysis Report] as important to safety is affected by these changes except for the low level CST HPCI/RCIC suction transfer function. Postulated malfunctions related to the proposed changes to the low level switches are bounded by the failure of the HPCI system, which has been previously evaluated in the USAR. The RCIC system is not relied upon to mitigate any USAR design basis accident.

No new types of credible events could be identified which could be created by the proposed setpoint change and level switch cross connection. No new failure modes are associated with the proposed changes [sic].

(3) The proposed amendment will not involve a significant reduction in the margin of safety.

No margin of safety is reduced. Technical Specification and other specified margins of safety are effectively increased by the proposed activities. The HPCI/RCIC low CST level suction transfer setpoint is being adjusted upward in the conservative direction. Cross connecting the level switches associated with this transfer will preserve the redundancy built into the logic during extended outages of one CST. A small additional reduction in the reliability of the automatic transfer logic due to possible freezing or plugging of common instrument piping results when the level switches are temporarily cross connected during infrequent periods of operation with one CST in service. This small reduction in reliability of the automatic transfer function is fully compensated for by the ability to perform necessary and prudent preventive maintenance on the CSTs while at the same time supplying the HPCI and RCIC systems with water from the preferred high quality source.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: Cynthia A. Carpenter.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota.

Date of amendment requests: November 25, 1998.

Description of amendment requests: The proposed amendments would modify the technical specifications (TS) (TS 3.2 and Table 3.5–2B) to allow limited inoperability of boric acid storage tank (BAST) level channels and transfer logic channels to provide for required testing and maintenance of the associated components.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment[s] will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not affect any system that is a contributor to initiating events for previously evaluated design basis accidents. Therefore, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated. The proposed Actions 34, 35 and 36 will allow limited continued plant operation with portions of BAST to RWST [refueling water storage tank] transfer instrumentation inoperable. However, because the proposed actions place time limits on inoperability comparable to those already approved for use in the Prairie Island Technical Specifications the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated. The remaining proposed changes to Table TS.3.5–2B and to Specification 3.2B are administrative in nature. The changes to Table 3.5–2B incorporate design information on the BAST to RWST transfer instrumentation which clarifies the operability requirements for the instrumentation. The changes to Specification 3.2.B add a reference to Table TS.3.5–2B. Therefore, because of the administrative nature of the changes, they do not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed changes do not alter the design or function of any plant component and do not install any new or different equipment. The proposed changes do not alter the operation of any plant component in a manner which could lead to a new or different kind of accident. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. The proposed amendment[s] will not involve a significant reduction in the margin of safety.

The proposed Actions 34, 35 and 36 will allow limited continued plant operation with portions of the BAST to RWST transfer instrumentation inoperable. However, because the proposed actions place time limits on inoperability comparable to those already approved for use in the Prairie Island Technical Specifications the proposed changes do not involve a significant reduction in the margin of safety. The remaining proposed changes to Table TS.3.5–2B and to Specification 3.2.B are administrative in nature. The changes to Table 3.5–2B incorporate design information on the BAST to RWST transfer instrumentation which clarifies the operability requirements for the instrumentation. The changes to Specification 3.2.B add a reference to Table TS.3.5–2B. Therefore, because of the administrative nature of the changes, they do not involve a significant reduction in the margin of safety.

NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: Cynthia A. Carpenter.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California.

Date of amendment request: September 3, 1998.

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to change TS 3.4.9.1, ''Reactor Coolant System—Pressure/Temperature Limits,'' Figure 3.4–2, ''Reactor Coolant System Heatup Limitations—Applicable Up to 12 EFPY,'' and Figure 3.4–3, ''Reactor Coolant System Cooldown Limitations—Applicable Up to 12 EFPY,'' to extend the applicability up to 16 effective full power years (EFPY). The affected TS Bases would also be appropriately revised.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to Figures 3.4–2 and 3.4–3 of Technical Specification (TS) 3.4.9.1 and the associated Bases adjust the reactor coolant system (RCS) heatup and cooldown pressure/temperature (P/T) limits to permit operation through 16 effective full power years (EFPY). The 16 EFPY P/T limits are more restrictive than the current limits; this accounts for an expected incremental increase in reactor vessel embrittlement, and assures the reactors will continue to be operated within acceptable stresses and at temperatures for which the reactor vessel metal exhibits ductile properties. The P/T limits developed for 16 EFPY were determined in accordance with 10 CFR 50, Appendix G, and maintain the same margins of safety as the current limits. The proposed changes will not impact the probability of overpressurization or brittle fracture of the vessel, and therefore will not impact the consequences of an accident.

The present low temperature overpressure protection (LTOP) pressure and enable temperature setpoints were reviewed and found to be acceptable and conservative for use through 16 EFPY, based on use of ASME Code Case N–514, which provides acceptable margins to the prevention of vessel overpressurization and brittle fracture. Therefore, there is no change to the consequences of accidents previously analyzed. Since no changes are proposed in the actual LTOP setpoints, nor any physical alteration of the LTOP system, nor a change to the method by which the LTOP system performs its function, there would be no change to the probability of an accident previously evaluated. The proposed change to the Bases incorporates use of ASME Code Case N–514, which will benefit DCPP by not resulting in a reduced RCS P/T window and reduced power-operated relief valve (PORV) pressure setpoint for LTOP. This maintains the current level of operator flexibility during heatup and cooldown, and prevents an increase in the probability of an accident associated with an inadvertent PORV actuation.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an 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 changes to TS 3.4.9.1, ''Reactor Coolant System—Pressure/Temperature Limits,'' do not involve any physical alteration to any plant system or change the method by which any safety-related system performs its function. The changes to TS 3.4.9.1 account for the effects of an incremental increase in reactor vessel embrittlement and are requested in order to restrict future reactor operation to within acceptable stress levels and temperature regimes in accordance with 10 CFR 50, Appendix G, requirements. These changes are needed to maintain the current P/T limit margins of safety as defined by 10 CFR 50, Appendix G, and ASME XI, Appendix G, for operation through 16 EFPY. The possibility of a new kind of accident such as catastrophic failure of the reactor vessel is prevented by maintaining acceptable margins of safety.

The present LTOP pressure setpoint was reviewed and found to be acceptable and conservative for the extension of the P/T curves to 16 EFPY.

Additionally, the proposed changes will not affect the ability of the LTOP system to provide pressure relief at low temperatures, thereby maintaining the LTOP design basis.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to TS 3.4.9.1 adjust the RCS heatup and cooldown P/T limits to permit operation through 16 EFPY. The P/T limits have been determined in accordance with 10 CFR 50, Appendix G, and include the safety margins with regard to brittle fracture required by the ASME Section XI, Appendix G, which maintain the same margins of safety as the current limits.

The LTOP setpoints were reevaluated using the requirements of ASME Code Case N–514. This code case was developed to provide the necessary margins of safety for the prevention of reactor vessel overpressurization and brittle fracture. The LTOP evaluation results conclude the current LTOP setpoints are conservative for operation through 16 EFPY. In addition, avoiding an unnecessary reduction in the LTOP, the PORV pressure setpoint prevents an increase in the likelihood of an inadvertent PORV actuation.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room Location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorney for Licensee: Christopher J. Warner, Esq., Pacific Gas & Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: William H. Bateman.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas.

Date of amendment request: October 29, 1998.

Description of amendment request: The proposed change will relocate Technical Specification 3/4.7.9 requirements for Snubbers and the associated Bases to the Technical Requirements Manual.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed change relocates requirements and surveillances for Technical Specification 3/4.7.9 that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The affected components are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected systems and components will be relocated from the Technical Specifications to the Technical Requirements Manual, which is incorporated in the STP UFSAR and will be maintained pursuant to 10 CFR 50.59. In addition, the Snubber operability is addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. The associated changes to the Index are administrative. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change relocates requirements and surveillances applicable to snubbers which does not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. The associated changes to the Index are administrative. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates requirements and surveillances for snubbers, that do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion in Technical Specifications. The change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component, or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety. The associated changes to the Index are administrative and have no potential effect on the margin of safety.

The proposed change is also consistent with the Westinghouse Plants Standard Technical Specification, NUREG–1431 approved by the NRC Staff, revising the Technical Specifications to reflect the approved content ensures no significant reduction in the margin of safety. Therefore, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas.

Date of amendment request: October 29, 1998.

Description of amendment request: The proposed change will relocate Specification 3/4.3.4, ''Turbine Overspeed Protection,'' to the Technical Requirements Manual.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed change relocates the requirements of specification 3/4.3.4, ''Turbine Overspeed Protection,'' that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The specification is not related to any assumed initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirement to perform the testing is not altered by the proposed change. The requirements of the limiting condition for operation and surveillance testing will be relocated from the Technical Specifications to the Technical Requirements Manual, which is incorporated in the STP UFSAR and will be maintained pursuant to 10 CFR 50.59. In addition, the surveillance testing details are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change relocates the requirements of specification 3/4.3.4, ''Turbine Overspeed Protection,'' that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates the requirements of specification 3/4.3.4, ''Turbine Overspeed Protection,'' that do not meet the 10 CFR 50.36 criteria for inclusion in Technical Specifications. The change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. In addition, the relocated requirements applicable to the turbine overspeed protection remain the same as the existing Technical Specifications requirements. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety.

The proposed change is also consistent with the Westinghouse Plants Standard Technical Specification, NUREG–1431 approved by the NRC Staff. Revising the Technical Specifications to reflect the approved content, ensures no significant reduction in the margin of safety. Therefore, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas.

Date of amendment request: October 29, 1998.

Description of amendment request: The proposed change will relocate descriptive details of Surveillance Requirement 4.8.1.1.2.g, regarding maintenance of the diesel generator fuel oil storage tanks (DGFOSTs), to the Technical Requirements Manual.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed change relocates descriptive details of surveillance requirement 4.8.1.1.2.g that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(3). The affected descriptive testing details are not related to any assumed initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirement to perform the testing is not altered by the proposed change. The descriptive details of the surveillance testing will be relocated from the Technical Specifications to the Technical Requirements Manual, which is incorporated in the STP UFSAR and will be maintained pursuant to 10 CFR 50.59. In addition, the surveillance testing details are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change relocates descriptive details of surveillance testing applicable to the DGFOSTs, which do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(3). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates descriptive details of the surveillance testing applicable to the DGFOSTs, that do not meet the 10 CFR 50.36 criteria for inclusion in Technical Specifications. The change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. In addition, the relocated surveillance testing details for the DGFOSTs remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety.

The proposed change is also consistent with the Westinghouse Plants (Improved) Standard Technical Specification, NUREG–1431, approved by the NRC Staff. Revising the Technical Specifications to reflect the approved NUREG–1431 content ensures no significant reduction in the margin of safety. Therefore, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri.

Date of application request: October 31, 1997, as supplemented by letter dated September 29, 1998. This notice supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that was published on January 14, 1998 (63 FR 2283).

Description of amendment request: The proposed amendment application would change Tables 3.3–3, 3.3–4, and 4.3–2 of the technical specifications (TS) to revise the engineered safety feature actuation system (ESFAS) Functional Unit 6.f, Loss of Offsite Power-Start Turbine-Driven Pump. Table 3.3–2 would be revised to create separate functional units for the analog and digital portions of the ESFAS function associated with starting the turbine-driven auxiliary feedwater pump (TDAFP) upon a loss of offsite power. Table 3.3–4 would be revised to create separate functional units for the analog and digital portions of the ESFAS function associated with starting the TDAFP upon a loss of offsite power. Table 4.3–2 would be revised to create separate functional units for the analog and digital portions of the ESFAS function associated with starting the TDAFP upon a loss of offsite power.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The recognition that different operability and surveillance requirements apply to analog vs. digital circuitry does not impact any previously analyzed accidents. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change does not alter the current method or procedures for meeting the surveillance requirements in Table 4.3–2. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an 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.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The separation of analog and digital portions of Functional Unit 6.f will not impact the normal method of plant operation.

The operability requirements, ACTION Statement, and surveillance requirements for the analog portion, new Functional Unit 6.f.1), are identical to those of Functional Unit 8.a. The requirements for the digital portion, new Functional Unit 6.f.2), are consistent with the current Technical Specifications, other than the new ACTION Statement 39 provisions that eliminate the transient imposed on the plant from a 3.0.3 shutdown and the performance of a refueling interval TADOT [Trip Actuating Device Operational Test]. There is no safety benefit associated with shutting the plant down under LCO 3.0.3, if both logic trains were inoperable, when considering the fact that the pump is allowed to be inoperable for 72 hours. This unnecessary shutdown would be detrimental to plant safety. The ''new'' TADOT requirement is a reflection of current plant testing practice. These changes do not change any ESFAS design standards and are appropriate for digital functions such as this. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does 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 change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: William H. Bateman.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri.

Date of application request: July 30, 1998.

Description of amendment request: The proposed amendment application would change Table 4.3–2 of the technical specifications (TS) by adding a table notation to clarify that verification of the time delays associated with engineered safety feature actuation system (ESFAS) Functional Units 8.a and 8.b, ''Loss of Power,'' is only performed as part of the channel calibration.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The protection systems will continue to function in a manner consistent with the plant design basis. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. Neither the Trip Setpoints and Allowable Values in Technical Specification Table 3.3–4 nor the response times listed in FSAR [Final Safety

Analysis Report] Table 16.3–2 are affected. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation capabilities. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an 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.

There are no hardware changes associated with this license amendment nor are there any changes in the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unchanged. Verification of the time delays need not be performed on a monthly basis when response time testing is performed on an alternating 18 month basis per the provisions of Technical Specifications 4.3.1.2 and 4.3.2.2 and the verification of LOCA [loss-of-coolant accident] and shutdown sequencer timing and analog channel time constant calibrations are performed on a refueling frequency. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change. Therefore, the proposed change does 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 change does not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR [Departure from Nucleate Boiling Ratio] limits, F_Q, Nuclear Enthalpy Rise Hot Channel Factor, LOCA PCT [Peak Clad Temperature], peak local power density, or any other margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: William H. Bateman.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339. North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia.

Date of amendment request: November 18, 1998.

Description of amendment request: The proposed amendments would make changes to the North Anna Power Station (NAPS), Unit 1 and 2, Technical Specifications (TS) Surveillance Requirement (SR) 4.7.13.1, ''Groundwater Surveillance Requirements'' and related Table 3.7–6, ''Allowable Groundwater Levels—Service Water Reservoir.'' The change in the SR requests that the measuring device numbers assigned to piezometers be eliminated from the TS SR in order to avoid redundancy, and eliminate confusion as well as the need to initiate TS changes whenever new piezometers are added, older devices are replaced or abandoned in-place. The proposed change in groundwater threshold levels will raise the allowable groundwater levels to those consistent with the allowable levels in the ''Stability of Service Water Reservoir (SWR) Slope Under Increased Phreatic Surface'' calculations.

Basis for proposed no significant hazards consideration determination: as required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards, which is presented below:

Specifically, operation of the North Anna Power Station in accordance with the proposed TS Change Request will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, since: (a) removing non-safety related SWR piezometer device numbers from the TS and raising TS allowable groundwater surface threshold elevation levels in the southeast section of the SWR will have no effect on the way the safety-related Service Water System was designed to operate, (b) Periodic Test Procedures will continue to identify all open-tube piezometers and require that they be monitored in order to obtain as much information as possible regarding changing groundwater levels, (c) sufficient redundancy will continue to exist since at least two (2) open-tube (standpipe-type) piezometers, not subject to mechanical failure, have been installed in each of the three (3) SWR zones to meet the TS Surveillance Requirement that ''at least one measurement per zone be available'' and (d) recent calculations have confirmed that raising the allowable water level in the southeast section of the SWR will not affect the stability of the SWR dike as indicated in the original design basis calculation.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, since: (a) the frequency of piezometer monitoring and the intent of monitoring groundwater surface threshold elevations in order to maintain stability of the SWR slope have not changed, (b) no physical modification to the plant or new mode of plant operation is involved, (c) changes are consistent with the assumptions made in the Safety Analyses and original design basis calculation and (d) failure of the SWR dike and ensuing loss of service water was the most serious accident postulated and considered credible. Operation of the SWR is not being changed. Therefore, a new or different kind of accident is [not] created by the change in groundwater level. In addition, since both the SWR and Lake Anna reservoir provide redundant sources of service water, failure of the SWR is not considered as a credible accident.

3. Involve a significant reduction in a margin safety, since: (a) increasing the allowable phreatic surface in the SE section of the SWR dike will not lower the factor of safety with respect to the stability of the SWR as defined by the original design basis calculation, (b) the margin to failure of the SWR dike has been proven by calculation to have not been reduced as defined by the original design basis calculation and (c) subject changes will not impact the performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions, therefore the margin of safety is not changed by the proposed [change] in groundwater level at the SWR.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for Licensee: Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: Herbert N. Berkow.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the

Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina.

Date of application for amendment: February 27, 1997, as supplemented August 24, 1998.

Brief description of amendment: This amendment changes Technical Specification (TS) 3/4.4.5, ''Steam Generators,'' by adding sleeve installation as an alternative to tube plugging for repairing degraded steam generators.

Date of issuance: November 23, 1998.

Effective date: November 23, 1998.

Amendment No.: 85.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17225).

The August 24, 1998, supplemental letter provided clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Commonwealth Edison Company, Docket No. 50–254, Quad Cities Nuclear Power Station, Unit 1, Rock Island County, Illinois.

Date of application for amendment: August 14, 1998, as supplemented by letters dated October 13 and November 23, 1998.

Brief description of amendment: The amendment changes the Quad Cities Technical Specifications (TS) to reflect the use of Siemens Power Corporation ATRIUM–9B fuel. Specifically the amendment incorporates the following into the TS: (a) new methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates, (b) administrative changes that eliminate the cycle specific implementation of ATRIUM–9B fuel and adopt Improved Standard Technical Specification language where appropriate, and (c) changes to the Minimum Critical Power Ratio.

Date of issuance: December 3, 1998.

Effective date: Immediately, to be implemented within 30 days.

Amendment No.: 182.

Facility Operating License No. DPR–29: The amendment revised the TSs. Public comments requested as to proposed no significant hazards consideration: Yes (63 FR 59588 dated November 4, 1998). This notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by December 4, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final no significant hazards consideration determination are contained in a Safety Evaluation dated December 3, 1998.

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida.

Date of application for amendment: October 31, 1997, as supplemented December 13, 1997, February 27 and April 24, 1998.

Brief description of amendment: The amendment proposed to revise the Final Safety Analysis Report (FSAR) to reflect changes to the credited methodology for boron precipitation prevention, as approved by the NRC.

Date of issuance: November 30, 1998.

Effective date: November 30, 1998.

Amendment No.: 171.

Facility Operating License No. DPR–72: Amendment revised the Operating License to reflect the change to the FSAR.

Date of initial notice in **Federal Register**: November 12, 1997 (62 FR 60731). The supplemental letters contained clarifying information that did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 30, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida.

Date of application for amendment: October 29, 1998.

Brief description of amendment: The amendment revised the terminology used in the St. Lucie Plant Technical Specifications (TS) relative to the implementation and automatic removal of certain protection system trip bypasses to ensure that the meaning of explicit terms used in the TS are consistent with the intent of the stated requirements.

Date of Issuance: November 24, 1998.

Effective Date: November 24, 1998.

Amendment No.: 98.

Facility Operating License No. NPF–16: Amendment revised the TS.

Date of initial notice in **Federal Register**: November 5, 1998 (63 FR 59809).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 24, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear

Generating Station, Ocean County, New Jersey.

Date of application for amendment: July 21, 1998.

Brief description of amendment: The amendment (1) revises Technical Specification (TS) 6.2.2.2(a) to provide flexibility to accommodate unexpected absence of on-duty shift crew members, (2) eliminates reference to the Manager, Plant Operations in Specification 6.2.2.2(j) as the position has been eliminated, (3) reduces the maximum time in which to forward audit reports to the responsible manager from 60 days to 30 days, (4) replaces the term ''Vice President'' with the term ''Corporate Officer'' in several places in Section 6, and (5) corrects several typographical errors.

Date of Issuance: November 30, 1998.

Effective date: November 30, 1998, to be implemented within 30 days

Amendment No: 203.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 26, 1998 (63 FR 45525).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 30, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan.

Date of application for amendments: October 8, 1998.

Brief description of amendments: The amendments would revise the Technical Specification Section 3.4.1.3, ''Reactor Coolant System—Shutdown,'' and its associated bases to provide separate requirements for the Reactor Coolant system in MODE 4, MODE 5 with the reactor coolant loops filled, and MODE 5 with the reactor coolant loops not filled.

Date of issuance: November 27, 1998.

Effective date: November 27, 1998, with full implementation within 30 days.

Amendment Nos.: 224 and 208.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 27, 1998 (63 FR 57322).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 27, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York.

Date of application for amendment: June 19, 1998, as supplemented November 6, 1998.

Brief description of amendment: This amendment changes Technical Specification 3.2.2 and the associated Bases to update pressure-temperature operating curves and tables for continued plant operation up to 28 effective full-power years.

Date of issuance: November 25, 1998.

Effective date: As of the date of issuance to be implemented before core operation exceeds 18 effective full-power years.

Amendment No.: 164.

Facility Operating License No. DPR–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40557)

The November 6, 1998, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 25, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York.

Date of application for amendment: November 25, 1998, as supplemented November 27, 1998.

Brief description of amendment: This change adds a note to certain specific containment isolation valves listed in Table 4.4–1. The note permits the licensee to operate Indian Point Unit 3 for the remainder of the current cycle (Cycle 10) without pneumatic leakage rate testing of these isolation valves. These valves have been leakage rate tested in the past using water pressurized with nitrogen gas. Without this emergency amendment, there would have had to delay its resumption of plant operation at power until the Technical Specifications required test was performed.

Date of issuance: November 27, 1998.

Effective date: As of the date of issuance to be implemented immediately.

Amendment No.: 184.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration, are contained in a Safety Evaluation dated November 27, 1998.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York.

Date of application for amendment: August 3, 1998, as supplemented October 20, 1998.

Brief description of amendment: The amendment provides for application of the existing minimum critical power ratio safety limit to Cycle 14 operation.

Date of issuance: November 25, 1998.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 246.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48264).

The October 20, 1998, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 25, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Southern California Edison Company, et al., Docket No. 50–362, San Onofre Nuclear Generating Station, Unit No. 3, San Diego County, California.

Date of application for amendment: September 22, 1998.

Brief description of amendment: The proposed amendment would modify the Technical Specifications (TS) to change the parameter used to establish and remove the bypasses for high reactor power trips. The parameter would be changed from the current ''THERMAL POWER'' to logarithmic power.

Date of issuance: November 23, 1998.

Effective date: November 23, 1998.

Amendment Nos.: 136.

Facility Operating License No. NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56259).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama.

Date of application for amendments: June 12 and August 14, 1998 (TS–390).

Brief description of amendments: Changes the technical specifications (TS) to accommodate surveillance intervals to be compatible with a 24-month fuel cycle.

Date of issuance: November 30, 1998.

Effective date: November 30, 1998.

Amendment Nos.: 235, 255, 215.

Facility Operating License Nos. DPR–33, DPR–52 and DPR–68: Amendments revised the TS.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48269).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 30, 1998.

No significant hazards consideration comments received: None.

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee.

Date of application for amendments: August 21, 1996 (TS 96–03).

Brief description of amendments: The amendments revise the SQN Technical Specification (TS) 3.7.1.3 to extend the limiting condition for operation of the condensate storage tanks to Mode 4 when steam generator is relied upon for heat removal.

Date of issuance: November 19, 1998.

Effective date: As of the date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: 238 and 228.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TSs.

Date of initial notice in **Federal Register**: October 9, 1996 (61 FR 52967).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 19, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee.

Date of application for amendments: April 30, 1998 (TS 98–01).

Brief description of amendments: The amendments revise the SQN Technical Specification Surveillance Requirement 4.4.3.2.1.b by changing the mode requirement to allow power-operated relief valve stroke testing in Modes 3, 4, and 5 with a steam bubble in the pressurizer rather than only in Mode 4.

Date of issuance: November 19, 1998.

Effective date: As of the date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: 239 and 229.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38204).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 19, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee.

Date of application for amendment: May 6, as supplemented June 5, 1998.

Brief description of amendment: The requested changes would allow an increase in the limit, up to 5.0 percent, for the U–235 enrichment of new (unirradiated) fuel stored in the new fuel storage racks and limit the fuel storage locations to assure that k-effective values are met.

Date of issuance: December 1, 1998.

Effective date: December 1, 1998.

Amendment No.: 15.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43214).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 1, 1998.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440 Perry Nuclear Power. Plant, Unit 1, Lake County, Ohio.

Date of application for amendment: September 3, 1998.

Brief description of amendment: This amendment revised Technical Specification 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' by increasing the Division 3 Diesel Generator fuel oil level requirements to account for (1) a rounding error in the calculation, and (2) the unusable volume due to vortex formation at the eductor suction nozzle located in the fuel oil storage tank.

Date of issuance: November 23, 1998.

Effective date: November 23, 1998.

Amendment No.: 94.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53960).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440 Perry Nuclear Power Plant, Unit 1, Lake County, Ohio.

Date of application for amendment: August 28, 1997.

Brief description of amendment: This amendment revised Pressure-

Temperature (P/T) Limits contained in Technical Specification 3.4.11 as a result of the Reactor Vessel Material Surveillance Program Requirements contained in Appendix H of 10 CFR Part 50.

Date of issuance: December 2, 1998.

Effective date: December 2, 1998.

Amendment No.: 95.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61846).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin.

Date of application for amendment: April 15, 1998 as supplemented by letters dated August 13, 1998, September 28, 1998, and November 24, 1998.

Brief description of amendment: The amendment incorporates changes to TS 2.1, ''Safety Limits'' and TS 3.10, ''Control Rod and Power Distribution Limits.'' These changes revise the power distribution peaking factor limits and limits operating parameters related to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) in support of cycle 23 fuel and reload changes. A change associated with the fuel and reload changes, is the removal, from the current licensing basis, of the fuel pool turbine missile hazards analysis

Date of issuance: December 2, 1998.

Effective date: December 2, 1998.

Amendment No.: 142.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 5, 1998 (63FR25120).

The supplemental submittals did not affect the initial determination of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

Dated at Rockville, Maryland, this 9th day of December 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 98–33206 Filed 12–15–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration (Boise Cascade Corporation, Common Stock, $2.50 Par Value; Associated Common Stock Purchase Rights); File No. 1–5057

December 10, 1998.

Boise Cascade Corporate (''Company'') has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') and Rule 12d2–2(d) promulgated thereunder, to withdraw the above specified securities (''Securities'') from listing and registration on the Pacific Exchange, Inc. (''PCX'' or ''Exchange'').

The reasons cited in the application for withdrawing the Securities from listing and registration include the following:

The Securities of the Company are currently listed on the New York Stock Exchange (''NYSE''), Chicago Stock Exchange (''CHX''), and PCX. The Company's Securities first traded on the PCX in 1965. Currently, the number of shares traded through the PCX is minimal, and has been declining over the last several years.

As part of an overall business review, the Company's management and Board of Directors considered the manner in which its stock is traded in the marketplace. The Company found the majority (well over 90%) of its Securities are traded on the NYSE. After considering many factors, the Company's management and Board of Directors determined that no significant business reasons exist for the Company to continue listing its Securities on the PCX. The Company intends to maintain its listing on the NYSE.

In compliance with the Exchange's rules, the Company sent the PCX a letter requesting voluntary delisting. The letter set out the basis for the Company's decision to delist, and provided a certified copy of the Board resolution authorizing this action.

On November 3, 1998, the Equity Listings Committee of the PCX approved the Company's request to be removed from listing and registration on the Exchange.

This application relates solely to the withdrawal from listing of the Company's Securities from the PCX and shall have no effect upon the continued listing of the Securities on the NYSE or the CHX.

Any interested person may, on or before January 4, 1999, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, N.W., Washington, D.C. 20549, facts bearing upon whether the application has been made in accordance with the rules of the Exchange and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

For the Commission, by the Division of Market Regulation, pursuant to delegated authority.

Jonathan G. Katz,

Secretary.

[FR Doc. 98–33301 Filed 12–15–98; 8:45 am]

BILLING CODE 8010–01–M

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting

''FEDERAL REGISTER'' CITATION OF PREVIOUS ANNOUNCEMENT: [To Be Published].

STATUS: Closed Meeting.

PLACE: 450 Fifth Street, NW., Washington, DC.

DATE PREVIOUSLY ANNOUNCED: To Be Published.

CHANGE IN THE MEETING: Date Change/ Time Change.

The closed meeting scheduled for Thursday, December 17, 1998, at 11:00 a.m., has been changed to Wednesday, December 16, 1998, at 2:00 p.m.

Commissioner Unger, as duty officer, determined that Commission business required the above change and that no earlier notice thereof was possible.

At times, changes in Commission priorities require alterations in the scheduling of meeting items. For further information and to ascertain what, if any, matters have been added, deleted or postponed, please contact:

The Office of the Secretary (202) 942–7070.