that are open to the public, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Richard K. Major, ACNW, as far in advance as practicable so that appropriate arrangements can be made to schedule the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting will be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for taking pictures may be obtained by contacting the ACNW office, prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should notify Mr. Major as to their particular needs.

In accordance with Subsection 10(d) Pub. L. 92–463, I have determined that it is necessary to close portions of this meeting noted above to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy per 5 U.S.C. 552b(c)(6)).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Richard K. Major, ACNW (Telephone 301/415–7366), between 8:00 A.M. and 5:00 P.M. EDT. ACNW meeting notices, meeting transcripts, and letter reports are now available for downloading or reviewing on the internet at http://www.nrc.gov/ACRSACNW.

Videoteleconferencing service is available for observing open sessions of ACNW meetings. Those wishing to use this service for observing ACNW meetings should contact Mr. Theron Brown, ACNW Audiovisual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m. EDT at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

The ACNW meeting dates for Calendar Year 2000 are provided below:

ACNW meeting No.	Meeting date
January 2000—No meeting.	
116th (Rockville, MD)	February 15–17, 2000.
117th (Rockville, MD)	March 14–16, 2000.
April 2000—No meeting.	
118th (Rockville, MD)	May 16–18, 2000.
119th (Rockville, MD)	June 20–22, 2000.
120th (San Antonio, Texas)	July 18–20, 2000.
August 2000—No meeting.	
121st (Amargosa Valley, Nevada)	September 19–21, 2000.
122nd (Rockville, MD)	October 17–19, 2000.
123rd (Rockville, MD)	November 15–17, 2000.
December 2000—No meeting.	

Dated: October 29, 1999.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 99–28786 Filed 11–2–99; 8:45 am]

BILLING CODE 7590–01–P

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. Public Law 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 October 8, 1999, through October 22, 1999. The last biweekly notice was published on October 20, 1999 (64 FR 56526).

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 6D59, 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 December 10, 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 electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room). 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 electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: September 28, 1999.

Description of amendment request: The licensee has proposed to revise Technical Specification (TS) 2.1.1, ''Reactor Core Safety Limits,'' and TS 5.6.5, ''Core Operating Limits Report.'' These revisions would remove cycle-specific safety limit restrictions which are no longer necessary.

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 license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The procedures for determining the MCPR [Minimum Critical Power Ratio] Safety Limit are described in General Electric Standard Application for Reactor Fuel (i.e., topical report NEDE–24011–P–A, otherwise referred to as GESTAR II). The basis for the MCPR Safety Limit calculation is to ensure that greater than 99.9 percent of all fuel rods in the core avoid transition boiling in the event of a postulated accident. The existing MCPR Safety Limit preserves this margin to transition boiling and fuel damage. The MCPR Safety Limits for the BSEP [Brunswick Steam Electric Plant], Unit 1 TSs, and their use in determining cycle-specific operating limits documented in the Core Operating Limits Report, are determined using NRC-approved methods (i.e., GESTAR II). The use of these methods ensures that the MCPR Safety Limit values are within the existing design and licensing bases, and cannot increase the probability or consequences of an accident previously evaluated.

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

The MCPR Safety Limit is a TS numerical value that has been established to ensure that fuel damage from transition boiling does not occur in at least 99.9 percent of the fuel rods in the core as a result of a limiting postulated accident. The MCPR Safety Limit is not an accident initiator; therefore, it cannot create the possibility of any new type of accident. The MCPR Safety Limits are calculated using NRC-approved methods. The function, location, operation, and handling of the fuel will remain unchanged. In addition, the initiating sequence of events for previously evaluated accidents has not been changed. Therefore, no new or different kind of accident has been created.

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

The MCPR Safety Limit preserves the existing margin to transition boiling and fuel damage in the event of a postulated accident. The margin of safety, as defined in the TS Bases, will remain the same. The MCPR Safety Limit remains unchanged, and will ensure that greater than 99.9 percent of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. The MCPR Safety Limits will continue to be calculated using NRC-approved generic and cycle-specific methodologies that are described in GESTAR II. Therefore, the proposed change does 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.

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Section Chief: Ron Hernan, Acting.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: September 28, 1999.

Description of amendment request: The amendment revises Technical Specifications (TS) surveillance requirement (SR) 3.7.6.2 ''Component Cooling Water (CCW) System,'' to change the CCW pump automatic start actuation signal basis from Engineered Safety Feature Actuation Signal (ESFAS) to Loss-of-Power Diesel Generator (LOP DG). This change is required to reflect the original plant design which was not properly incorporated during conversion of the TS to Improved TS.

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:

Carolina Power & Light (CP&L) Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. The CP&L conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change to Surveillance Requirement (SR) 3.7.6.2 does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The safety function of the Loss of Power (LOP) Diesel Generator (DG) start signal for the Component Cooling Water (CCW) pumps is to start the CCW pumps in order to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment. The CCW pumps start upon receipt of a LOP DG start signal from undervoltage on the emergency bus. The LOP DG start signal to the CCW pumps is not an Engineered Safety Features Actuation System (ESFAS) signal. Since this proposed change only corrects the description of the start signal, the proposed change does not involve an 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 does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change corrects the word description of the start signal for the CCW pumps and does not alter any plant design margin or analysis assumption as described in the Updated Safety Analysis Report. The proposed change does not affect any limiting safety system setpoint, calibration method, or setpoint calculation. Therefore, the proposed change does not involve a 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.

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602 .

NRC Section Chief: Sheri R. Peterson.

CBS Corporation (licensee), Westinghouse Test Reactor, Waltz Mill Site, Westmoreland, Pennsylvania, Docket No. 50–22, License No. TR–2

Date of amendment request: September 15, 1999, as supplemented on October 4, 1999.

Description of amendment request: CBS Corporation 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 revise four Technical Specifications (TS) in their approved Decommissioning Plan. The first TS change deals with what doors need to be closed when restricted activities are taking place within containment. Access to containment is through three locations, i.e., the truck lock door and the east and west airlock doors. Each entry point has two doors, an outer door and an inner door. In the existing TS either door could be closed except during personnel ingress or egress or while equipment is being passed through the doorways. In the proposed TS the licensee has specified the following. For the truck lock door the inner door to containment needs to be closed. The reason given for the change is that the containment boundary is more accurately defined as the interior access door between the truck lock area and containment. The truck lock area was transferred to the SNM–770 license in April 1970 and the outer doors are controlled by this license.

For the east and west airlock doors, fire doors with an interior crash bar have been installed at the outer door as a safety feature to minimize the risk of personnel being trapped in containment during an emergency. The airlock doors (inner doors) do not allow quick and efficient egress during a postulated fire in containment; therefore, the original air lock doors have been removed and confinement is maintained by the newly installed fire doors.

Therefore the proposed TS require that the inner truck lock door be closed and the outer east and west lock doors be closed except during personnel ingress or egress or while equipment is being passed through the doorways, and this meets the original goal of the existing TS.

The second TS change deals with the condition of the containment when the containment is open for removal of materials and equipment. In the existing TS Restricted Activities in containment are suspended. In the proposed TS, containment extension is permitted if an enclosure is provided around the opening to effectively isolate the containment from the outside environment. If these extensions are not in place, all Restricted Activities in containment are suspended. Negative pressure (airflow into containment) is maintained in containment in the existing as well as the proposed TS. Containment isolation is effectively maintained under the proposed TS as it was in the existing TS.

The third TS change deals with the control of access into containment. In the existing TS the outer doors in the air lock and the truck lock outer doors shall be locked or blocked closed to prevent unauthorized entry except when authorized personnel are inside the containment building or outside with the door in view. In the proposed TS access into containment is through a Health Physics (HP) control point, which is on the first floor of the G-Building. To prevent unauthorized entry the accesses into and out of containment shall be locked or blocked closed except when this access control point is supervised and the provisions of the first TS change are implemented.

Normal access to the containment is through a door in the G-Building basement (east and west airlock doors). The G-Building basement is a ‘‘Radiation Area’’. Routine activities during the day may require workers to exit containment (rest, lunch, equipment change out, etc). Locking or blocking the doors after workers temporarily exit during the working day does not minimize radiation dose and reduces worker efficiency. Access control will be established on the first floor of the G-Building outside the radiation area. Therefore, the access control point would provide positive control into and out of containment and meets the original intent of the TS.

The fourth TS is being changed to include the HP control point in the monthly visual surveillance, which assures that accesses into containment are locked or blocked when no on is inside containment and the HP control point is not occupied.

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 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 the margin of safety.

The staff agrees with the licensee’s no significant hazards consideration determination submitted on September 15, 1999, for the following reason:

The changes are consistent with the original intent of the TS, i.e., to maintain confinement during Restricted Activities and to prevent uncontrolled spread of contamination. Access control is still being maintained.

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 licensee: William David Wall, Assistant General Counsel, CBS Corporation, 11 Stanwix Street, Pittsburgh, Pennsylvania 15222.

NRC Branch Chief: Ledyard B. Marsh.

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: June 2, 1999, as supplemented August 25, 1999.

Description of amendment request: The proposed amendment would relocate the quality assurance (QA) related requirements to the licensee’s Quality Assurance Program Description (QAPD) in accordance with NRC Administrative Letter (AL) 95–06, ‘‘Relocation of Technical Specifications Administrative Controls Related to Quality Assurance,’’ dated December 12, 1995. Specifically, Technical Specification (TS) Section 6.5, ‘‘Review and Audit,’’ TS Section 6.8, ‘‘Procedures and Programs,’’ and TS Section 6.10, ‘‘Record Retention’’ would be relocated from the current TS to the QAPD in accordance with 10 CFR 50.36 (60 FR 30957).

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 proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: This amendment application does not involve a significant increase in the probability or consequences of an accident previously analyzed. The relocation of the administrative controls from the Technical Specification to the Quality Assurance Program Description (QAPD) does not alter the performance or frequency of these activities. Any future changes to the QA Program Description, which might constitute a reduction in commitments, are governed by 10 CFR 50.54(a). Therefore, sufficient controls for these requirements exist and these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response: This amendment application does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes involve the relocation of requirements from the Technical Specifications to the QAPD.

Relocation of these requirements does not affect plant equipment or the way the plant operates. The functions continue to be performed in the identical manner as they are currently being performed. Therefore, the proposed revisions can not create a new or different kind of accident.

3. Does the proposed license amendment involve a significant reduction in a margin of safety?

Response: This amendment application does not involve a significant reduction in a margin of safety. The requested Technical Specification revisions relocate the administrative control requirements from the Technical Specifications to the QAPD. These requirements are not being altered by this relocation. The functions continue to be performed in the identical manned as they are currently being performed. Any future changes to the QA Program Description, which might constitute a reduction in commitments, are governed by 10 CFR 50.54(a). Therefore, sufficient controls for these requirements exist and these 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 request involves no significant hazards consideration.

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

NRC Section Chief: Sheri Peterson.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: July 30, 1999 (NRC–99–0048).

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to include provisions related to enabling the oscillation power range monitor (OPRM) upscale trip function in the average power range monitor. This change is associated with the power range neutron monitoring (PRNM) system installed during the last refueling outage. The associated Bases would also be 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 change is to enable the OPRM Upscale Function that is contained in the previously installed PRNM equipment. Enabling the OPRM hardware provides the long-term stability solution required by Generic Letter 94–02. This hardware incorporates the Option III detect and suppress solution reviewed and approved by the NRC in the Reference 6, 7, and 8 [of the licensee's application dated July 30, 1999] Licensing Topical Reports and their Supplements. The OPRM is designed to meet all requirements of GDC [General Design Criteria] 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic power oscillations prior to violating the fuel MCPR [minimum critical power ratio] Safety Limit. The OPRM system provides this protection in the region where Interim Corrective Actions (ICAs) restricted operation because of stability concerns. Thus, the ICA restrictions on plant operation are deleted from the TS, including region avoidance and the requirement for the operator to manually scram the reactor with no recirculation loops operating. Operation at high core powers with low core flows may cause a slight, but not significant, increase in the probability that an instability may occur. This slight increase is acceptable because subsequent to the automatic detection of an instability, the OPRM Upscale function provides an automatic scram signal to the RPS that is faster than the operator-initiated manual scram required by the current ICAs. Because of this rapid automatic action, the consequences of an instability event are not increased as a result of the installation of the OPRM system because it eliminates dependence on operator actions.

Based on the above discussion, 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.

The proposed change permits Fermi 2 to enable the OPRM power oscillation detect and suppress function provided in previously installed PRNM hardware, and it simultaneously deletes certain restrictions which preclude operation in regions of the power-flow map where oscillations potentially may occur. Enabling the OPRM Upscale function does not create any new system hardware interfaces nor create any new system interactions. Potential failures of the OPRM Upscale function result either in failure to perform a mitigation action or in spurious initiation of a reactor scram. These failures would not create the possibility of a new or different kind of accident.

Based on the above discussion, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The OPRM Upscale function implements BWROG [Boiling Water Reactor Owners Group] Stability Option III, which was developed to meet the requirements of GDC 10 and GDC 12 by providing a hardware system that detects the presence of thermal-hydraulic instabilities and automatically initiates the necessary actions to suppress the oscillations prior to violating the MCPR Safety Limit. The NRC has reviewed and accepted the Option III methodology described in the Reference 6, 7, and 8 [of the licensee's application dated July 30, 1999] Licensing Topical Reports and their supplements, and concluded that this solution will provide the intended protection. Therefore, it is concluded that there will be no reduction in the margin of safety as defined in the TS as a result of enabling the OPRM Upscale function and simultaneously removing the operating restrictions previously imposed by the ICAs.

Based on the above discussion, the proposed 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 amendment request involves no significant hazards consideration.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: Claudia M. Craig.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: September 10, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirements (SRs) 3.8.4.1, 3.8.4.6, and 3.8.6.2 to accommodate changes in battery parameters associated with the replacement of the Division I battery. The licensee also plans to revise the Bases section for SR 3.8.6.2.

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

The proposed changes do not involve a change in the manner in which the plant is operated. TS Sections [SRs] 3.8.4.1, 3.8.4.6, 3.8.6.2 and Bases Surveillance Requirement Section 3.8.6.2 are being revised to reflect the new Division I battery cell/system characteristics and associated requirements. The new battery will have an increased capacity over the present battery, while maintaining the existing battery system voltage requirements. This is possible because the present and new battery specific gravity (1.215) and type (lead calcium) are the same. Also, the end of battery system discharge voltage remains the same as 210 VDC. The Division I batteries will continue to furnish power to redundant essential loads as required and as designed. The new surveillance requirement voltages are based on the same volts/cell criteria used for the existing batteries. Furthermore, failure or malfunction of the station batteries does not initiate any of the analyzed accidents previously evaluated in the UFSAR [updated final safety analysis report]. The changes described will therefore not involve an increase in the probability or consequences of an accident previously evaluated.

2. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The new battery is Class 1E qualified equipment and is being maintained within the same overall design parameters as the existing battery. That is, the battery terminal voltage on float voltage conditions (2.167 volt[s]/cell), overvoltage conditions (2.5 volts/cell) and charger capability (2.15 volts/cell) are the same as the original design. Furthermore, the end of system discharge voltage of the battery system is maintained the same; therefore, there is no negative impact to plant loads supplied by the batteries. Failures of the batteries and chargers have been considered in both the existing and modified configurations. The proposed changes will not change performance or reliability nor introduce any new or different failure modes or common mode failure and will therefore not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The changes do not involve a significant reduction in the margin of safety.

The changes act to increase overall battery capacity from 560 ampere-hours to 1200 ampere-hours with the minimum battery discharge voltage remaining at 210 VDC (or 105 VDC per battery). The battery terminal voltage on float voltage conditions (2.167 volt[s]/cell), overvoltage conditions (2.5 volts/cell) and charger capability (2.15 volts/cell) are the same as the original design. The new surveillance requirement voltages are based on the same volts/cell criteria used for the existing batteries. The batteries' ability to satisfy the design requirements (battery duty cycle) of the dc system will not be reduced from original plant design and will therefore not have any negative impact to plant loads [that] the battery supplies. The proposed changes therefore do not involve 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 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 licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: Claudia M. Craig.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: April 5, 1999; supplemented October 7, 1999.

Description of amendment request: The proposed amendments would revise the Improved Technical Specifications (TS), Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report to incorporate Topical Report (TR) DPC-NE–3005–P, ''Thermal-Hydraulic Transient Analysis Methodology.'' The proposed changes are: (1) Modification of a note for TS Surveillance Requirement (SR) 3.4.1.2, ''RCS [Reactor Coolant System] Pressure, Temperature, and Flow DNB [Departure from Nucleate Boiling] Limits,'' to add that the SR would apply for the condition where there is a 0°F delta-Tcold setpoint; (2) modification of TS 3.4.10, ''Pressurizer Safety Valves,'' to increase the setpoint range of the lift settings for the pressurizer safety valves; (3) modification of SR 3.4.10.1 to specify that the pressurizer safety valve lift settings shall be within plus or minus 1 percent; (4) addition of TS 3.7.4, ''Atmospheric Dump Valve (ADV) Flow Paths,'' to address the applicability and required actions related to the ADS valves; (5) addition of TS 3.9.7, ''Unborated Water Source Isolation Valves,'' to require valves that are used to isolate unborated water sources to be secured in the closed position while in Mode 6, provide required actions if one or more of the valves is not secured in the closed position, and related SRs; (6) TS 5.6.5b would be changed to update the Core Operating Limits Report references; and (7) modification of the appropriate Bases to reflect the above changes and consistentcy with the revision to the TR analysis. In addition, proposed changes to the UFSAR revisions were provided.

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

No. The proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) incorporate the accident analyses established in Topical Report DPC–NE–3005–P, ''UFSAR Chapter 15 Transient Analysis Methodology, Revision 1.'' On February 1, 1999, Duke submitted Topical Report DPC–NE–3005–P to the NRC for approval. The NRC found DPC–NE–3005–P acceptable as noted in SER [Safety Evaluation Report] dated May 25, 1999.

The analyzed events are initiated by the failure of specific plant structures, systems or components. These proposed changes do not impact the condition or performance of those structures, systems or components.

The revised accident analyses in DPC–NE–3005–P demonstrate that the applicable acceptance criteria are met. In addition, the calculations show that the applicable radiological and environmental acceptance criteria will continue to be met.

Based on the above, the proposed changes do 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?

No. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses demonstrate that the applicable acceptance criteria are met. As a result, no new failure modes are being introduced.

Based on the above, the proposed changes do not create the possibility of any new or different kind of accident from any accident previously evaluated.

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

No. The margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses in DPC–NE–3005–P demonstrate that the applicable acceptance criteria are met.

Based on the above, 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 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 licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

NRC Section Chief: Richard L. Emch, Jr.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: September 9, 1999.

Description of amendment request: The proposed amendment would increase the authorized rated thermal power level of 3579 megawatts thermal by 5 percent to 3758 megawatts thermal. The proposal follows the NRC-approved generic format and content for Boiling Water Reactor power uprate licensing topical reports documented in NEDC–31897P–A, ''Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,'' and NEDC–31984P, ''Generic Evaluations of General Electric Boiling Water Reactor Power Uprate.''

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

The increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level, as the regulatory criteria established for plant equipment (ASME code, IEEE standards, NEMA standards, Regulatory Guide criteria, etc.) are still complied with at the uprated power level. An evaluation of the boiling water reactor (BWR) probabilistic risk assessments concludes that the calculated core damage frequencies do not significantly change due to power uprate. Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to uprate. No new challenge to safety-related equipment results from power uprate.

The changes in consequences of hypothetical accidents which would occur from 102% of the uprated power, compared to those previously evaluated from greater than or equal to 102% of the original power, are in all cases insignificant, because the accident evaluations from power uprate compared with 105% of original power do not result in exceeding the NRC-approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met at the uprated power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in NEDO–24011, ''GESTAR II.'' Challenges to fuel (ECCS performance) are evaluated, and shown to still meet the criteria of 10 CFR 50.46 and Appendix K (Section 4.3 above, and Regulatory Guide 1.70 Safety Analysis Report Section 6.3).

Challenges to the containment have been evaluated, and the containment and its associated cooling systems will continue to meet 10 CFR Appendix A Criterion 38, Long Term Cooling, and Criterion 50, Containment.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 100 (Regulatory Guide 1.70 Safety Analysis Report Chapter 15).

(2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

As summarized below, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Equipment that could be affected by power uprate has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode was identified. The full spectrum of accident considerations defined in Regulatory Guide 1.70 has been evaluated and no new or different kind of accident has been identified. Power uprate uses existing technology, and applies it within the capabilities of already existing plant equipment in accordance with existing regulatory criteria and includes NRC approved codes, standards, and methods. General Electric has designed BWRs of higher power and no new power dependent accidents have been identified.

The technical specifications needed to implement power uprate require some small adjustments, with no change to the plant's physical configuration. All technical specification changes have been evaluated and are acceptable.

(3) Will the change involve a significant reduction in a margin of safety?

As summarized below, this change will not involve a significant reduction in a margin of safety.

The calculated loads on all affected structures, systems and components remain within their design allowables for all design basis event categories. No NRC acceptance criteria are exceeded. Some design and operational margins are affected by power uprate, however, the margins of safety originally designed into the plant are not affected by power uprate. Because the plant configuration and reactions to transients and hypothetical accidents do not exceed the presently approved NRC acceptance limits, power uprate does 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.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: September 9, 1999.

Description of amendment request: The proposed amendment would revise Perry Operating License Appendix B, the Perry Environmental Protection Plan. The proposed change will eliminate the requirement in the Environmental Protection Plan to sample Lake Erie sediment in the Perry and Eastlake Plant area for *Corbicula*, since *Corbicula* and zebra mussels have already been identified, and control and treatment plans have been implemented which are effective on both species.

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 Perry Plant water source (Lake Erie) is now known to have mussels and clams present. Therefore, it is no longer necessary to use lake sampling techniques designed to provide advance notice of their arrival. Treatment programs and monitoring for system fouling are in place. The treatment programs and system monitoring for fouling makes it highly likely that equipment degradation due to *Corbicula* would be avoided or readily identified, allowing time for corrective actions. Therefore, the programs will ensure that plant systems remain capable of performing their intended functions. Since the lake sampling was designed to allow time to implement a control program, and the control program is now in place, elimination of the lake sampling program will not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

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

The proposed change will eliminate the lake sampling program designed to detect the arrival of *Corbicula*, a particular species of clam, at the Perry Plant. Since the clam is now known to exist in the vicinity, and control methods are developed and implemented, advanced detection is no longer required. Since the proposed change involves only a monitoring program and does not change or modify the design, maintenance or operation of any plant equipment, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The current requirements for aquatic monitoring are designed to detect *Corbicula* prior to plant cooling water systems and heat exchangers becoming infested with clams and flow becoming degraded, and thus reducing the cooling available to safety systems.

Since an effective control method has already been implemented, the deletion of a lake sampling method to provide advance warning of clams in the area provides no significant benefit. The proposed change will continue to provide the same level of protection against system or component fouling that currently exists, thus the proposed change will 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 amendment request involves no significant hazards consideration.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308

NRC Section Chief: Anthony J. Mendiola.

First Energy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: September 9, 1999.

Description of amendment request: The proposed amendment includes nine separate changes to the Perry technical specifications. The proposed changes include increasing the minimum water volume of the condensate storage tank, clarification of minimum ECCS pump differential pressures, clarifications to Required Action and Condition statements, as well as minor nomenclature and editorial changes.

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

A summary of the proposed changes is:

1. (Condensate Storage Tank (CST) Level-Low.) The Allowable Values for the CST low water level limits (Technical Specification (TS) Table 3.3.5.1–1 Function 3.d and Table 3.3.5.2–1 Function 3) are being revised from greater than or equal to 59,700 gallons to greater than or equal to 90,300 gallons based on recent revisions to calculations taking into account potential vortex issues. This change also results in raising the TS Surveillance Requirement (SR) 3.5.2.2.b value for the normal CST level limit to greater than or equal to 249,700 gallons.

2. (Emergency Core Cooling System Pump Differential Pressure) TS SRs 3.5.1.4 and SR 3.5.2.5 are being revised to better describe what the differential pressures listed in the SRs represent at Perry Nuclear Power Plant, in lieu of the phrase ''pump differential pressure'.

3. (RCIC/RHR Steam Line Flow-High) The proposed change revises the nomenclature on a table to match the plant-specific instrument nomenclature.

4. (Containment Average Temperature-To-Relative-Humidity) This revision is a clarification to prevent misinterpretation of the Required Actions.

5. (Containment Vacuum Breakers) T 3.6.1.11 Required Action A.2 is being revised to clarify the proper actions to take if the required number of vacuum breakers is not operable. Required Action A.2 is being revised to add the word ''required'.

6. (Reporting Requirements) TS Administrative Controls Reporting Requirement 5.6.1 is being revised to clarify the definition of the time period of the report. ''Calendar'' is being removed from the term ''calendar year'' to clarify the time period that the Occupational Radiation Exposure Report is required to cover, to be consistent with the revised wording in 10 CFR 20.1003.

7. (High Radiation Area) TS Administrative Control 5.7 is being revised to update the titles of individuals responsible for radiation protection. The term ''health physics'' is being revised to ''radiation protection'' to be consistent with plant terminology.

8. (ECCS Instrumentation) Required Action E.1 Note 1 is being revised for consistency with other specifications. The word ''in'' is being added.

9. (Electrical Power Systems) In TS 3.8.3, the word ''continued'' is being added to the bottom of the page for consistency with other specifications.

The CST level change is adjusted in a conservative direction, as recommended by NRC inspectors during a Safety System Functional Inspection (SSFI) that was conducted in the spring of 1997. The current setpoints were reviewed and determined to be adequate, however it was suggested that some additional margin should be added. The ''low level'' limits are being raised to move the setpoint further away from the level at which vortexing would begin, and the normal water level limit is also being raised to ensure that at least 150,000 gallons of water would be available for HPCS and RCIC. Since the existing limits are already considered adequate, and the proposed changes are in the conservative direction, the proposed change does not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

The other eight proposed changes are administrative only, and can have no effect on any previously evaluated accident scenario. These eight changes have no effect on plant hardware, plant design, safety limit settings, or system operation and therefore do not modify or add any initiating parameters that would significantly increase the probability of an accident previously evaluated, or the radiological consequences of an event.

(2) The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will raise the Condensate Storage Tank level, which is conservative, and also includes some administrative changes to improve clarity, update titles or terminology. None of these changes can create the possibility of a new of different kind of accident from any accident previously evaluated.

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

The Condensate Storage Tank level change increases the margin of safety by providing more margin between the setpoint that causes the HPCS and RCIC suctions to shift from the CST to the Suppression Pool and the beginning of the formation of a vortex at their pump suctions. The other administrative changes have no effect on the margin of safety. Therefore the proposed change will 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 amendment request involves no significant hazards consideration.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: September 14, 1999.

Description of amendment request: The proposed amendment would delete one Operating License Condition, and revise another. License Condition 2.C.10 regarding controls over the containment air locks during plant outages would be deleted due to the effective implementation of Shutdown Safety administrative controls at Perry. License Condition 2.F would be revised to clarify the intent of reporting requirements for violations of the technical specifications and the Environmental Protection Plan.

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 delete or revise two Operating License Conditions, one that addresses administrative controls on air locks during refueling outages, and one regarding reporting of violations of the technical specifications and the Environmental Protection Plan.

These proposed changes to the Operating License are administrative only, and have no effect on any previously evaluated accident scenario. The proposed changes have no effect on plant hardware, plant design, safety limit setting, or plant system operation and therefore do not modify or add any initiating parameters that would significantly increase the probability of an accident previously evaluated.

The changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients, nor will they alter the operation of equipment important to safety previously evaluated in the accident analyses.

The proposed activity does not affect accident mitigation capabilities or the radiation release amounts for postulated accidents. Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged.

Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes are administrative in nature, and do not involve any physical alteration of the plant (no new or different type of equipment will be installed). They do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes have no impact on component and system interactions.

The safety functions of plant structures, systems, and components are also not changed in any manner, nor is the reliability of any structure, system, or component reduced.

The proposed changes are not providing for operation in a mode that is not already evaluated. These changes do not affect the operation of any systems or components, nor do they involve any potential initiating events that would create any new or different kind of event.

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 will not involve a significant reduction in the margin of safety.

The proposed changes are administrative in nature (they delete or revise two license conditions). Administrative controls will continue to be applied to the opening of the air locks during plant shutdown periods, and to the reporting of violations of the technical specifications and the Environmental Protection Plan.

There is no impact on safety limits or limiting safety system settings. The changes do not affect any plant safety parameters or setpoints. No physical or operational changes to the facility will result from the proposed changes.

The proposed changes have no impact on any safety analysis assumptions. Consequently, no margin of safety as described in the Final Safety Analysis Report or defined in the basis of any technical specification is reduced as a result of these changes. These proposed changes do not detrimentally affect the ability of structures, systems, and components important to safety to fulfill their intended safety functions.

Therefore, the proposed changes do not cause 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 amendment request involves no significant hazards consideration.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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 amendment requests: October 12, 1999.

Description of amendment requests: The proposed amendments would revise Technical Specification (T/S) Surveillance Requirement (SR) 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the associated T/S Bases. Additionally, certain administrative text format changes are being proposed.

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 of occurrence or consequences of an accident previously evaluated?

The proposed changes relocate the details associated with the acceptance criteria and test parameters from the T/S SR to the associated Bases and do not affect system operability or performance. The format changes in the text on each page are administrative in nature and do not result in any change in plant operation. Relocation of this information to the Bases is administrative in nature and does not affect the probability or consequences of any accident previously evaluated. No actual change to the requirement is made. Actual plant operation is not affected by the administrative changes. No methods of operation of plant systems, structures or components are changed. Operation of accident mitigation features is not changed. Consequently, there is no affect upon the probability of any previously analyzed accident, transient, accident initiators, or precursor events. Additionally, because there is no actual change in plant design or operation, there is no affect upon radioactive material inventories, plant shielding, or effluent release points. Therefore, these changes do not significantly increase the probability of occurrence 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 changes relocate the details associated with the acceptance criteria and test parameters from the T/S SR to the associated Bases and do not affect system operability or performance. The format changes in the text on each page are administrative in nature and do not result in any change in plant operation. Facility operation and procedures are not changed. Relocation of this information to the Bases is administrative in nature and does not affect [sic] create any new accident scenarios, accident initiators, or precursor events. Therefore, the proposed changes do 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 proposed changes relocate the details associated with acceptance criteria and test parameters from the T/S SR to the associated Bases and do not modify T/S safety settings, setpoints, or other values. The format changes in the text on each page are administrative in nature and do not result in any change in plant operation. There is no effect upon operating margins and accident margins because the administrative changes do not change the manner of operation of plant systems, structures, or components. Plant emergency and abnormal operating procedures are not affected. There is no change of actual testing methodology, test parameters, or acceptance criteria. The response of the plant to an event is the same. Potential offsite doses are unaffected because operation of the facility is unchanged. Relocation of the testing details to the Bases is acceptable because controls are in place for T/S Bases changes which require evaluation of changes under the provisions of 10 CFR 50.59. 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Jeremy J. Euto, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: September 30, 1999.

Description of amendment request: The proposed amendment would change the Technical Specification surveillance periodicity requirements for the control room emergency filtration system.

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.

During an accident, the Control Room Emergency Filtration [EFT] System provides filtered air to pressurize the Control Room to minimize the activity, and therefore the radiological dose, inside the Control Room. Technical Specification surveillance requirements are established in order to ensure that the EFT System will perform its safety function during an accident. The proposed amendment eliminates unnecessary testing which is not required to show that the filters are operable and which causes unnecessary wear and tear on the system. The remaining surveillances adequately show that the system is operable and capable of performing its safety function. Dose to the public and the Control Room operators are not affected by the proposed change.

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, the proposed amendment will not significantly increase the probability or the consequences of an accident previously evaluated.

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

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes. The proposed surveillance requirements are consistent with industry and regulatory guidance and show that the system is capable of performing its safety function. System reliability is enhanced by the proposed change by eliminating unnecessary wear on the system.

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

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

The proposed amendment is within current industry and regulatory standards for testing filters. The proposed amendment maintains margins of safety. Off-site and Control Room dose assessments are not affected by the proposed amendment, since the ability of the EFT System to perform its safety function is shown by the proposed surveillance requirements. The proposed change to the surveillance provides assurance that the system will perform at the filter efficiency used in the evaluation of the radiological consequences of the postulated events. Therefore, the proposed amendment will 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 amendment request involves no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: September 30, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications associated with the Safety Limit Minimum Critical Power Ratios (SLMCPRs) in order to support the operation of Hope Creek in the upcoming Cycle 10 with a mixed core of General Electric (GE) and Asea Brown Bovieri/Combustion Engineering (ABB/CE) fuel. In addition, administrative changes would be made to the Technical Specifications to reflect the change in fuel vendor from GE to ABB/CE.

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

The derivation of the revised SLMCPRs for Hope Creek for incorporation into the Technical Specifications, and its use to determine cycle-specific thermal limits, have been performed using NRC [U.S. Nuclear Regulatory Commission] approved methods. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

There are no significant increases in the consequences of an accident previously evaluated. The basis of the MCPR Safety Limit is to ensure that no mechanistic fuel damage due to clad overheating is calculated to occur if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased.

Removal of the cycle specific footnote for the Safety Limit applicability will not involve a significant increase in the probability or consequences of an accident previously evaluated since the change is administrative and does not affect the plant or fuel design or operation.

Likewise, the proposed changes to the Average Planar Heat Generation Rate (APLHGR), Minimum Critical Power Ratio (MCPR), Recirculation Loop Limiting Condition for Operation (LCO) Action Statements, and references to fuel vendor analyses and reports do not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes to the APLHGR, MCPR and Recirculation Loop LCOs are considered to be administrative in nature since the Core Operating Limits Report (COLR) will continue to be used to appropriately control and limit the bounds of plant operation with slow control rods or during single recirculation loop operation, and the COLR will still be developed in accordance with NRC approved methods. Similarly, the revised references to the fuel vendor throughout the Technical Specifications are also considered to be administrative in nature since they reflect the current status of NRC approval of methodologies utilized by PSE&G [Public Service Electric and Gas Company] and the fuel vendor to develop operating and safety limits for the fuel and core designs. These proposed changes do not alter the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

There are no significant increases in the consequences of an accident previously evaluated. The basis of the COLR and the PSE&G and fuel vendor methodologies is to ensure that no mechanistic fuel damage is calculated to occur if the limits on plant operation are not violated. The COLR will continue to preserve the existing margin to fuel damage and the probability of fuel damage is not increased.

Therefore, the proposed change does not involve an 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 contained in this submittal result from an analysis of the reload core using the same fuel types as previous cycles and an ABB/CE fuel design with extensive operating experience. These changes do not involve any new method for operating the facility and do not involve any facility modifications for the reload core operation. No new initiating events or transients result from these changes. Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident, from any accident previously evaluated.

Removal of the cycle specific footnote for the Safety Limit applicability does not create the possibility of a new or different kind of accident from any accident previously evaluated since the change is administrative and does not affect the plant or fuel design or operation.

The changes to the APLHGR, MCPR and Recirculation Loop LCOs are considered to be administrative in nature since the Core Operating Limits Report (COLR) will continue to be used to appropriately control and limit the bounds of plant operation with slow control rods or during single recirculation loop operation, and the COLR will still be developed in accordance with NRC approved methods. These changes do not involve any new method for operating the facility and do not involve any facility modifications in addition to the new fuel design. No new initiating events or transients result from these changes. Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident.

The revised references to the fuel vendor throughout the Technical Specifications are also considered to be administrative in nature since they reflect the current status of NRC approval of methodologies utilized by PSE&G and the fuel vendor to develop operating and safety limits for the fuel and core designs. These changes do not involve any new method for operating the facility and do not involve any facility modifications in addition to the new fuel design. No new initiating events or transients result from these changes. Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident.

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

The margin of safety as defined in the Technical Specification bases will remain the same. The new SLMCPRs are calculated using NRC approved methods, which are in accordance with the current fuel designs, and licensing criteria. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed Technical Specification changes do not involve a significant reduction in a margin of safety.

Removal of the cycle specific footnote for the Safety Limit applicability does not create the possibility of a new or different kind of accident from any accident previously evaluated since the SLMCPR will continue to be evaluated on a cycle-specific basis.

The margin of safety as defined in the Technical Specification bases will likewise remain unaffected by the proposed changes to APLHGR, MCPR and Recirculation Loop LCOs, and the revised references to the fuel vendor throughout the Technical Specifications. These changes establish controls for plant operation and establish bases for fuel analyses that reflect NRC approved methods, and are in accordance with the current fuel design and licensing criteria. These changes will continue to ensure that the plant is operated within specified acceptable fuel design limits. Therefore, the proposed Technical Specification 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 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 licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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: September 8, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3/4.8.1, ''A.C. Sources, Operating,'' and associated Bases, by eliminating the requirement for accelerated testing of the standby diesel generators and the associated reporting requirements. The TS Index would also be revised to reflect these changes.

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 changes do not involve hardware changes nor do they affect the operational limits or design of the standby diesel generators or power systems. These changes do not alter assumptions made in the safety analysis. In conjunction with the maintenance rule program, these changes continue to assure the operability and reliability of the standby diesel generators while minimizing the number of required engine starts and associated wear. These changes are also consistent with the guidance provided in Generic Letter 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators.''

Therefore, the proposed changes do not involve an 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 changes minimize the number of required standby diesel generator starts; they do not affect the operational limits or design. The performance capability of the standby diesel generators is not affected. These changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. These changes do not alter assumptions made in the safety analysis. These changes are also consistent with the guidance provided in Generic Letter 94–01.

Therefore, the changes will 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 proposed changes do not involve a change in the operational limits or design of the emergency power system. The design and capabilities of the standby diesel generators are not affected by these changes. These changes are also consistent with the guidance provided in Generic Letter 94–01.

The proposed changes 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 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.

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

NRC Section Chief: Robert A. Gramm.

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: September 8, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification 3/4.8.1, ''A.C. Sources, Operating,'' and associated Bases, by relocating the 18-month surveillance to subject the standby diesel generator to inspections in accordance with procedures prepared in conjunction with its manufacturer's recommendations, to the Updated Final Safety Analysis Report.

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 moves the requirement to perform manufacturer's recommended inspections of the Standby Diesel Generators from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR). The change does not result in any hardware or operating procedure changes. The requirement being removed from the Technical Specifications is not the initiator of any analyzed event. The UFSAR is maintained using the provisions of 10 CFR 50.59. Since any changes will be evaluated per 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC approval. Therefore, the changes do 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 moves the requirement to perform manufacturer's recommended inspections of the Standby Diesel Generators from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR). The change does not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. The change does not impose different requirements. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change will 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 moves the requirement to perform manufacturer's recommended inspections of the Standby Diesel Generators from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR). The change does not reduce the margin of safety since the location of details has no impact on any safety analysis assumptions. In addition, the requirement being transposed from the Technical Specification to the UFSAR [is the] same as the existing Technical Specification. Also, the UFSAR is maintained using the provisions of 10 CFR 50.59. Since any changes will be evaluated per 10 CFR 50.59, no significant reduction in a margin of safety will be allowed without prior NRC approval.

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.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 28, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications to increase the maximum allowable leakage rates for main steam isolation valves.

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

TVA proposes to utilize the main steam drain lines to preferentially direct MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this approach, the main steam lines, steam drain piping, and the main condenser are used to mitigate the consequences of an accident to limit potential off-site exposures below those specified in 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19 for control room dose limits.

Seismic verification walkdowns and evaluations of representative piping/supports were performed to demonstrate the main steam line piping and components that comprise the ALT path were rugged, and able to perform the safety function of MSIV leakage control following an Design Basis Earthquake (DBE). Thus, it has been concluded the primary components in the MSIV alternate treatment flow path can be relied upon to maintain structural integrity.

Therefore, the proposed amendment does not involve changes to structures, components, or systems which would affect the probability of an accident previously evaluated in the Browns Ferry Final Safety Analysis Report (FSAR).

A plant-specific radiological analysis has been performed to assess the effects of the proposed increase in MSIV leakage criteria in terms of off-site doses and main control room dose. This analysis uses the holdup and plate-out factors described in NEDC–31858P, Revision 2. The analysis shows the dose contribution from the proposed increase in leakage criteria is acceptable compared to doses limits prescribed in 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19. Therefore, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

B. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes require the use of the main steam piping and the condenser to process MSIV leakage. This additional function does not compromise the reliability of these systems. They will continue to function as intended and not be subject to a failure of a different kind than previously considered. In addition, MSIV functionality will not be adversely impacted by the increased leakage limit. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change to TS Surveillance Requirement 3.6.1.3.10 to increase the allowable MSIV leakage does not involve a significant reduction in the margin of safety. The allowable leak rate specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass containment. The results of the re-analysis supporting these changes were evaluated against the dose limits contained in 10 CFR 100 for off-site doses and 10 CFR 50, Appendix A, GDC 19 for control room doses. Sufficient margin relative to the regulatory limits is maintained even when conservative assumptions and methods are utilized. Therefore, the proposed change does 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.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: September 30, 1999.

Description of amendment request: The proposed amendments consist of administrative revisions to the Operating Licenses for BFN Units 1, 2 and 3 that delete license conditions that have become outdated, are no longer applicable, or are redundant, and consolidate license conditions which currently exist in two locations in each units' Technical Specifications.

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

The changes requested by this submittal are administrative in nature and do not change the way BFN operates. The proposed changes are intended to: delete redundant paragraphs, delete requirements and authorizations for modifications that have been completed, delete an authorization to temporarily store radioactive material on site, delete an exemption from a General Design Criterion which has expired, and consolidate license conditions which currently exist in two locations in each units Technical Specifications.

The change does not affect any design bases accident or the ability of any safe shutdown equipment to perform its design function. There are no physical modifications that are required to implement this license condition update. There is no impact on plant equipment or changes to operating procedures. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes described above are administrative in nature and do not change the way BFN operates. There are no physical modifications authorized by the proposed changes and there are no procedure or process changes that are requested. Changes requested are intended to ensure the license conditions reflect the current status of the plant. There is no impact on any accident analysis created by this change. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The changes described above are administrative in nature and do not change the way BFN operates. There are no procedural or physical changes required by this amendment. The license conditions are being updated partially as a result of NRC Information Notice 97–43 which highlighted the importance of periodically verifying compliance with the Operating License. These changes are intended to delete license conditions which are no longer needed or are redundant in order to ensure the license conditions accurately reflect the current status of the licensed facility. The change does not affect any design bases accident or the ability of any safe shutdown equipment to perform its design function, therefore no margins of safety have been affected by any of these changes. Accordingly, the proposed amendment does 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.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Acting Section Chief: Ronald W. Hernan.

Previously Published Notices of Consideration of Issuance of Amendment to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content of the same as above. They were published as individual notices either because the time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards considerations.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendment requests: June 8, 1999.

Brief description of amendments request: The proposed amendments would revise Technical Specification (TS) 3.7.15, ''Fuel Storage Pool Boron Concentration,'' TS 3.7.17, ''Spent Fuel Assembly Storage,'' and TS 4.3.1, ''Criticality,'' to increase spent fuel pool storage capacity by crediting soluble boron and decay time in the safety analysis for the spent fuel pool storage racks. The proposed amendments would also increase the maximum radially averaged fuel enrichment from 4.3 weight percent to 4.8 weight percent.

Date of publication of individual notice in **Federal Register***:* September 20, 1999 (64 FR 50835)

Expiration date of individual notice: October 20, 1999.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendment requests: October 8, 1999.

Brief description of amendments request: The proposed amendment would revise Technical Specification (TS) Section 3.8.4, ''DC Sources— Operating,'' to waive, on a one-time basis, the requirement to perform Surveillance Requirement (SR) 3.8.4.8 for Unit 1 channels A, B, and C.

Date of publication of individual notice in **Federal Register***:* October 19, 1999 (64 FR 56369).

Expiration date of individual notice: For comments on proposed no significant hazards consideration determination: November 2, 1999; for opportunity for hearing: November 18, 1999.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment request: October 20, 1998 (PCN 485), as supplemented August 13, 1999.

Brief description of amendment request: The proposed amendments would revise the San Onofre Nuclear Generating Station Units 2 and 3 technical specifications Surveillance Requirement 3.3.9 to include a response time testing requirement for the control room isolation signal.

Date of publication of individual notice in **Federal Register***:* October 12, 1999 (64 FR 55311.

Expiration date of individual notice: November 12, 1999.

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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendment request: October 27, 1998.

Brief description of amendment: The amendments update the Operating Licenses for the Brunswick Steam Electric Plant, Units 1 and 2.

Date of issuance: October 5, 1999.

Effective date: October 5, 1999.

Amendment No.: 206 and 236.

Facility Operating License Nos. DPR–71 and DPR–62: Amendment revises the Operating Licenses.

Date of initial notice in **Federal Register***:* December 30, 1998 (63 FR 71964).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 1999.

No significant hazards consideration comments received: No.

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: June 2, 1999, as supplemented on September 1, 1999.

Brief description of amendment: This amendment relocates Technical Specification (TS) Section 6.5, ''REVIEW AND AUDIT,'' TS 6.8.2, TS 6.8.3, and TS Section 6.10, ''RECORD RETENTION,'' intact from the Harris Nuclear Plant (HNP) TS to the Quality Assurance Program Description (QAPD) currently located in HNP Final Safety Analysis Report Section 17.3. Future changes to the associated relocated TS will be processed in accordance with 10 CFR 50.54(a). The change is consistent with NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995, and with the guidance provided in NRC Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls related To Quality Assurance,'' dated December 12, 1995.

Date of issuance: October 19, 1999.

Effective date: October 19, 1999.

Amendment No.: 92.

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

Date of initial notice in **Federal Register***:* June 30, 1999 (64 FR 35201).

The September 1, 1999, submittal contained 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 October 19, 1999.

No significant hazards consideration comments received: No.

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 application for amendments: June 30, 1999.

Brief description of amendments: The amendments revised the requirements related to the cross-tie of DC power buses between units, remove references to the AT&T batteries which have been replaced at Braidwood Station, and remove references to the 10-day allowed outage time (AOT) required for replacement of the AT&T batteries at Braidwood, Unit 2, which was granted in Amendment Nos. 99 and 99 issued to Braidwood Station, Unit Nos. 1 and 2, on March 26, 1999.

Date of issuance: October 13, 1999.

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

Amendment Nos.: 111 and 104.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register***:* August 11, 1999 (64 FR 43767).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 13, 1999.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: November 9, 1998, and July 7, 1999.

Brief description of amendments: The amendments revised Technical Specification Table 3.3.3–2, ''Emergency Core Cooling System Actuation Instrumentation Setpoints,'' to modify the degraded voltage second level undervoltage relay setpoint and allowable value.

Date of issuance: October 15, 1999.

Effective date: Immediately, to be implemented prior to startup from L1R08 for Unit 1 and prior to startup from L2R08 for Unit 2.

Amendment Nos.: 135 and 120.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register***:* January 13, 1999 (64 FR 2245) and August 11, 1999 (64 FR 43769).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 15, 1999.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: August 13, 1999, as supplemented on August 27, 1999.

Brief description of amendments: The amendments revise Technical Specification (TS) Section 1.0, ''Definitions,'' Item 1.7, ''Core Alteration,'' to specify that instrumentation and control rod movements are not considered core alterations if there are no fuel assemblies in the associated cell. The amendments also revise TS Sections 3/4.1, 3/4.3, and 3/4.9 to reflect the change in definition. In addition, a license condition is added as follows: ''The licensee is prohibited from moving any fuel assemblies within the reactor pressure vessel unless all control rods except one are fully inserted during refueling in Mode 5''.

Date of issuance: October 18, 1999.

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

Amendment Nos.: 136 and 121.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Operating Licenses and Technical Specifications.

Date of initial notice in **Federal Register**: September 8, 1999 (64 FR 48860).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 18, 1999.

No significant hazards consideration comments received: No.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of application of amendment: June 3, 1999, and as supplemented by letter dated August 24, 1999 .

Brief description of amendment: The amendment revises the Operating License to clarify that the license is not terminated until the Commission notifies the licensee in writing, and relocates certain Technical Specification (TS) requirements to licensee-controlled documents. The administrative controls section of the TSs have been revised to more closely conform to the standardized TSs. Administrative controls have been added for the control of radioactive effluents. A TS Bases Control Program has been added. The weight limit for loads carried over the spent fuel pool (SFP) has been increased. The amendment deletes certain TSs that are either (1) no longer applicable to the permanently shutdown and defueled state of the reactor, or (2) which duplicate regulatory requirements, or (3) which duplicate information located in the Updated Final Safety Analysis Report. A number of editorial changes were made to clarify the language used, to correct typographical errors, to renumber the listings, to remove section numbers that no longer contain requirements, and to renumber the pages in the TSs.

Date of issuance: October 19, 1999.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 195.

Facility Operating License No. DPR–61: The amendment revised the Operating License and the Technical Specifications.

Date of original notice in **Federal Register:** July 14, 1999 (64 FR 38024).

The August 24, 1999, supplement contained clarifications of the June 3, 1999 amendment request. The supplemental information did not change the staff's initial proposed no significant hazards consideration determination nor expand the scope of the original notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 1999.

No significant hazards consideration received: No.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: September 24, 1999.

Description of amendment request: The amendment revises current Technical Specification (TS) 3.6.1.8 by adding footnote ''**'' to Action b. The footnote allows continued operation of Fermi 2 with the leakage of penetration X–26 exceeding the limit in TS 4.6.1.8.2, provided certain compensatory measures are taken. Operation is allowed to continue until the next plant shutdown.

Because the NRC staff issued the Fermi 2 improved standard TSs (ITS) on September 30, 1999, with implementation within 90 days, this amendment also provides pages that are compatible with the ITS. The amendment adds a new special operations TS, ITS 3.10.8, to address the compensatory actions and other requirements associated with penetration X–26.

Date of issuance: October 19, 1999.

Effective date: October 19, 1999, and shall be implemented within 5 days.

Amendment No.: 135.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (64 FR 53421, dated October 1, 1999). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by November 1, 1999, but indicated that if the Commission makes a final NSHC 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 NSHC determination are contained in a Safety Evaluation dated October 19, 1999.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: Claudia M. Craig.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: June 23, 1999, as supplemented by letters dated August 6, September 8, and October 4, 1999.

Brief description of amendment: The amendment revises Technical Specification requirements for handling irradiated fuel in the Containment Building and in the Auxiliary Building, and selected specifications associated with performing core alterations.

Date of issuance: October 20, 1999.

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

Amendment No: 139.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications and Operating License.

Date of initial notice in **Federal Register**: August 25, 1999 (64 FR 46435).

The August 6, September 8, and October 4, 1999, submittals provided additional clarifying information and did not change the initial proposed no significant hazards consideration determination and did not expand the scope of the original application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 8, 1999.

Brief description of amendments: The amendments revised the Technical Specifications (TS), Section 6.0, Administrative Controls, by removing requirements that are adequately controlled by existing regulations other than 10 CFR 50.36 and the TS. The amendments also relocate selected requirements from TS 6.0 to licensee-controlled documents or programs (e.g., the final safety analysis report or the quality assurance plan). Guidance on the changes was developed by the NRC and provided in the Standard Technical Specifications for Pressurized Water Reactor Plants, NUREG–1431, and Administrative Letter 95–06, ''Relocation of Technical Specification

Administrative Controls Related to Quality Assurance,'' issued on December 12, 1995.

Date of issuance: October 6, 1999.

Effective date: As of date of issue, to be implemented within 90 days of issuance.

Amendment Nos.: 201 and 195.

Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the TS.

Date of initial notice in **Federal Register***:* April 7, 1999 (64 FR 17025).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 6, 1999.

No significant hazards consideration comments received: No.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: May 10, 1999, as supplemented July 16 and October 4, 1999.

Brief description of amendment: The amendment revised Duane Arnold Energy Center (DAEC) Technical Specification (TS) 2.1.1.2 to revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) to support operation with GE–12 fuel with a 10x10 pin array.

Date of issuance: October 20, 1999.

Effective date: Immediately, to be implemented within 30 days

Amendment No.: 229.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* July 14, 1999 (64 FR 38029).

The July 16 and October 4, 1999, letters provided additional clarifying information within the scope of the original **Federal Register** notice and did not affect the NRC staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 1999.

No significant hazards consideration comments received: No.

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: September 14, 1998.

Brief description of amendments: The amendments revise Technical Specification page 3/4 5–6, ''Limiting Conditions for Operation and Surveillance Requirements—Emergency Core Cooling Systems (ECCS),'' and its associated Bases to change pump runout limits for a safety injection pump to 675 gallons per minute (gpm) unless the pump is specifically tested to a higher flow rate not to exceed 700 gpm for Units 1 and 2.

Date of issuance: October 21, 1999.

Effective date: October 21, 1999, with full implementation within 45 days.

Amendment Nos.: 229 and 212.

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

Date of initial notice in **Federal Register:** August 31, 1999 (64 FR 47533).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 21, 1999.

No significant hazards consideration comments received: No.

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: May 21, 1999.

Brief description of amendments: The amendments change the Technical Specifications (TS) to allow reactor coolant system temperature changes in certain Mode 5 and 6 action statements if the shutdown margin is sufficient to accommodate the expected temperature change. In addition, footnotes regarding additions of water from the refueling water storage tank to the reactor coolant system are clarified and relocated to action statements. Additional actions are added in Table 3.3–1, ''Reactor Trip System Instrumentation,'' when the required source range neutron flux channel is inoperable. Corresponding changes are proposed for the Bases for TS 3/4.1.1, ''Boration Control,'' and TS 3/4.1.2, ''Boration Systems.'' Administrative changes are proposed to improve clarity. Finally, additions are made to shutdown margin TS surveillance requirements to address use of a boron penalty (requirement for additional boron) during residual heat removal system operation in Modes 4 and 5.

Date of issuance: October 21, 1999.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 230 and 213.

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

Date of initial notice in **Federal Register:** July 12, 1999 (64 FR 37574).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 21, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 31, 1998, as supplemented May 17, 1999.

Brief description of amendment: The amendment revises the technical specification reactor pressure vessel (RPV) pressure-temperature limit curves, deletes completed RPV sample surveillance requirements, deletes the requirement to withdraw a specimen at the next refueling outage, removes the standby liquid control system relief valve setpoint, and makes associated administrative changes.

Date of issuance: October 12, 1999.

Effective date: October 12, 1999, with full implementation within 45 days.

Amendment No.: 106.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 10, 1999 (64 FR 6706). The May 17, 1999, submittal added clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 1999.

No significant hazards consideration comments received: No.

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: April 9, 1999.

Brief description of amendment: The amendment changes the Technical Specifications by increasing the allowable outage time for any one safety injection pump.

Date of issuance: October 12, 1999.

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

Amendment No.: 196.

Facility Operating License No. DPR–64: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register****:* June 2, 1999 (64 FR 297147).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 1999.

No significant hazards consideration comments received: No.

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: January 29, 1999, as supplemented August 2, 1999.

Brief description of amendment: The amendment changes the Technical Specifications by increasing the allowable control rod misalignment when operating at or below 85% power.

Date of issuance: October 14, 1999.

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

Amendment No.: 197.

Facility Operating License No. DPR–64: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register***: April 21, 1999 (64 FR 19564).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 14, 1999.

No significant hazards consideration comments received: No.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: August 19, 1999, as supplemented by letter dated October 8, 1999.

Brief description of amendment: The amendment revises the TS to incorporate the new Pressure/Temperature Limits Curves consistent with the analysis results of reactor specimen W.

Date of issuance: October 21, 1999.

Effective date: October 21, 1999.

Amendment No.: 143.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register***: September 8, 1999 (64 FR 48865). The October 8, 1999, submittal contained 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 October 21, 1999.

No significant hazards consideration comments received: No.

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: March 2, 1999 (TS 98–05).

Brief description of amendments: The amendments delete the Sequoyah Nuclear Plant. License Conditions that require an Independent Safety Engineering Group.

Date of issuance: October 12, 1999.

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

Amendment Nos.: 248 and 239.

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

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24201).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 1999.

No significant hazards consideration comments received: No.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: August 18, 1999.

Brief description of amendment: The amendment revises the definition of "Surveillance Frequency" to incorporate provisions that apply upon the discovery of a missed Technical Specification surveillance. This change allows a delay in performing the actions of the associated limiting conditions for operation for up to 24 hours or up to the limit of the specified frequency, whichever is less, when it is discovered that a surveillance was not performed within its specified frequency.

Date of Issuance: October 13, 1999.

Effective date: October 13, 1999, and shall be implemented within 30 days.

Amendment No.: 179.

Facility Operating License No. DPR–28. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1999 (64 FR 48867).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 13, 1999.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: July 8, 1999, as supplemented by letter dated September 2, 1999.

Brief description of amendment: The amendment increased the allowable values for engineered safety features actuation system (ESFAS) loss-of-power 4 kV undervoltage trips in the current Technical Specifications (TSs) Table 3.3–4 (functional units 8.a and 8.b) and in surveillance requirement (SR) 3.3.5.3 of the improved TSs. The word "nominal" is also added to describe the trip setpoint in SR 3.3.5.3 and in the Bases of the improved TSs. The improved TSs were issued in Amendment 123 dated March 31, 1999, but have not yet been implemented.

Date of issuance: October 12, 1999.

Effective date: October 12, 1999, to be implemented within 60 days from the date of issuance.

Amendment No.: 128.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 11, 1999 (64 FR 43782).

The September 2, 1999, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 1999.

No significant hazards consideration comments received: No.

Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of application for amendment: March 17, 1999

Brief description of amendment: Revises the Possession Only License by deleting technical specifications related to hours of work and putting these requirements in appropriate Administrative Procedures.

Date of issuance: October 8, 1999.

Effective date: October 8, 1999, Implementation of this amendment includes incorporation of hours of work restrictions into the Administrative Procedures as described in the licensee's application dated March 17, 1999, and evaluated in the staff's safety evaluation attached to the amendment, and written notification to NRC that the amendment has been fully implemented.

Amendment No.: 153.

Facility Operating License No. DPR–3. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 7, 1999 (64 FR 17032).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 8, 1999.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 27th day of October 1999.

For the Nuclear Regulatory Commission.

Suzanne C. Black,

Deputy Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–28598 Filed 11–2–99; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–220 and 50–410]

Niagara Mohawk Power Corporation Nine Mile Point Nuclear Station, unit Nos. 1 and 2 Issuance of Final Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has taken action with regard to a letter dated April 5, 1999, (Petition) filed by Robert Norway (Petitioner) pursuant to § 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206). The Petitioner requested that the U.S. Nuclear Regulatory Commission (Commission or NRC) take action with regard to Niagara Mohawk Power Corporation (NMPC) and its senior nuclear and corporate management. The Petitioner requested that the Commission (1) take enforcement action against NMPC and its senior nuclear and corporate management and, as a minimum, against three named individuals, for submitting an altered 1994 employee record to the NRC at a predecisional enforcement conference on May 10, 1996; (2) take enforcement action against these same parties for presenting at this predecisional enforcement conference a false written record of what the Administrative Law Judge determined in the Department of Labor's proceeding in 95–ERA–005; (3) take enforcement action against these same parties for placing confidential employee information into the public record in violation of 10 CFR 2.790; and (4) take enforcement action against these same parties for an additional act of discrimination, pursuant to 10 CFR 50.7, for destroying the Petitioner's credibility and reputation in the nuclear industry. The Petitioner also requested that the NRC forward these issues to the Department of Justice for consideration of criminal prosecution.

In addition to these requests for enforcement actions, the Petitioner also requested that the following other actions be implemented: (1) That the agency perform an independent review of all of NMPC's docketed files associated with the individuals who committed the alleged fraud; (2) that the NRC forward the complaint to the NRC's Office of the Inspector General for an investigation of possible deliberate misconduct on the part of the NRC staff; (3) that an independent oversight group be established to oversee the NMPC Human Resources Department and Employee Concerns Program; (4) that a public meeting be held to obtain public comments pertaining to a number of issues, including discrimination and the placement of fraudulent documentation into public records; and (5) that the NRC publicly post NMPC's Safety Evaluation 96–09, which addresses the Residual Heat Removal Alternate Shutdown Cooling for Unit 2, to make it available for public comment, or require NMPC to re-perform this safety evaluation.

The Director of the Office of Nuclear Reactor Regulation has complied with the Petitioner's request to have his complaint forwarded to the NRC's Office of the Inspector General. The Petitioner's technical concern has been addressed independent of the Director's Decision by the NRC staff's letter to the Petitioner dated October 6, 1999. The Petitioner's additional requests are not supported for the reasons that are explained in the ''Final Director's Decision Pursuant to 10 CFR 2.206'' (DD–99–13). The complete text of the Final Director's Decision follows this notice and is available for public inspection at the Commission's Public Document Rooms located in the Gelman Building, 2120 L Street, NW., Washington, DC, and in the Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

A copy of the Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c) of the Commission's regulations. As provided for by this regulation, the Decision will constitute the final action of the Commission 25 days after the date of issuance of the Decision unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 28th day of October 1999.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 99–28759 Filed 11–2–99; 8:15 am]

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NUCLEAR REGULATORY COMMISSION

Notice of Availability of Draft Revision To NUREG–1574; Standard Review Plan for Antitrust Reviews

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability: Draft Revision 1 to Nureg 1574, ''Standard Review Plan (SRP) for Antitrust Reviews''.

SUMMARY: The NRC is seeking public comment on a Draft Revision to NUREG–1574, ''Standard Review Plan on Antitrust Reviews.'' The Standard Review Plan (SRP) is being revised in accordance with Commission guidance to remove any implication that the NRC would conduct antitrust reviews of license transfers after issuance of an operating license. The draft revised SRP is being published to obtain public comments which will be considered in evaluating whether the NRC review process in this area should be changed. The revised draft SRP will be available on NRC electronic bulletin boards and in the NRC's Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC 20555–001. A free single copy of Draft Revision 1 to NUREG–1574, to the extent of supply, may be requested by writing to U.S. Nuclear Regulatory Commission, Records Management Branch, Washington, DC 20555–0001.

DATES: The public is invited to submit comments on the revised draft SRP by January 3, 2000. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date. On the basis of the submitted comments, the Commission will determine whether to modify the revised draft SRP before issuing it in final form.

ADDRESSES: Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch.

Deliver comments to: 11555 Rockville Pike, Rockville, Maryland, between 7:45 a.m. and 4:15 p.m., Federal workdays.

SUPPLEMENTARY INFORMATION: The Draft Revision to NUREG–1574, ''Standard Review Plan on Antitrust Reviews,'' describes the procedures used by the NRC staff to implement the antitrust review and enforcement prescribed in Sections 105 and 186 of the Atomic Energy Act of 1954, as amended, and will replace the final NUREG–1574 published in December 1997. These