Organizational Research and Development
Steve Floyd—Nuclear Energy Institute
David Garchow—PSEG Nuclea
Richard Hill—Southern Nuclear Operating Company
Rod Krich—Exelon Corporation
Robert Laurie—California Energy Commission
James Moorman, III—U.S. Nuclear Regulatory Commission
Loren Plisco—U.S. Nuclear Regulatory Commission
Steven Reynolds—U.S. Nuclear Regulatory Commission
A. Edward Scherer—Southern California Edison Company
James Setser—Georgia Department of Natural Resources
Raymond Shadis—New England Coalition on Nuclear Pollution
James Trapp—U.S. Nuclear Regulatory Commission

A tentative agenda of the meeting is outlined as follows:

April 2, 2001

9:00 a.m. Introduction/Meeting Objectives and Goals/Review of Meeting Minutes from February 26–27, 2001 Meeting
9:30 a.m. Update from NRC Staff on the Reactor Oversight Process—Bill Dean/NRR
—Self-Assessment Program
—Results of the Internal/External Lessons Learned Workshops
12:15 p.m. Lunch
1:15 p.m. IIEP Members Feedback from the Reactor Oversight Process Lessons Learned Workshop
2:00 p.m. Presentations by Invited Stakeholders
3:00 p.m. Discussion of Consensus on Final List of Issues
4:00 p.m. Panel Discussion of Narrative Developed in Support of IIEP Issues
6:00 p.m. Adjourn

April 3, 2001 Meeting

8:00 a.m. Recap of Previous Day's Meeting/Meeting Objectives and Goals
8:30 a.m. Panel Discussion of Narrative Developed in Support of IIEP Issues
12:00 p.m. Lunch
1:00 p.m. Panel Discussion of Narrative Developed in Support of IIEP Issues
2:00 p.m. Agenda Planning Session/ Public Comments/General Discussion
3:00 p.m. Adjourn

Meetings of the IIEP are open to the members of the public. Oral or written views may be presented by the members of the public, including members of the nuclear industry. Persons desiring to make oral statements should notify Mr. Loren R. Plisco (Telephone 404/562–4501, e-mail LRP@nrc.gov) or Mr. John D. Monninger (Telephone 301/415–3495, e-mail JDM@nrc.gov) five days prior to the meeting date, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras will be permitted during this meeting.

Further information regarding topics of discussion; whether the meeting has been canceled, rescheduled, or relocated; and the Panel Chairman's ruling regarding requests to present oral statements and time allotted, may be obtained by contacting Mr. Loren R. Plisco or Mr. John D. Monninger between 8:00 a.m. and 4:30 p.m. EST.

IIEP meeting transcripts and meeting reports will be available from the Commission's Public Document Room. Transcripts will be placed on the agency's web page.

Dated: March 15, 2001.

Andrew Bates,

Advisory Committee Management Officer.

[FR Doc. 01–6985 Filed 3–20–01; 8:45 a.m.]

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 February 26 through March 9, 2001. The last biweekly notice was published on March 7, 2001 (66 FR 13797).

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 Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public

Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By April 20, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible 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: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible 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 application for amendments: April 26, 2000, as supplemented November 6, 2000. This notice supersedes the notice concerning this facility that appeared at 65 FR 31356, May 17, 2000.

Description of amendments request: The proposed amendments would revise the maximum Ultimate Heat Sink (UHS) temperature allowed by Technical Specification (TS) 3.7.2, ''Service Water (SW) System and Ultimate Heat Sink (UHS),'' for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The maximum 24-hour average UHS temperature specified in Required Action H.1 would be revised from 89°F to 90.5°F. To provide consistency with the new maximum 24-hour average UHS temperature, these amendments would also: (1) Revise the Condition H temperature range from ''>89°F and ≤92°F'' to ''>90.5°F and ≤92°F''; and (2) revise Surveillance Requirement 3.7.2.2 to require verification that the UHS temperature is ≤90.5°F versus ≤89°F.

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

1. Operation with the maximum 24 hour average UHS water temperature as high as 90.5°F does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The BSEP SW system is designed to provide cooling water for the removal of heat from equipment required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. This equipment includes the Diesel Generators (DGs), Residual Heat Removal (RHR) pump seal coolers, room cooling units for Emergency Core Cooling System (ECCS) equipment, and Residual Heat Removal Service Water (RHRSW) heat exchangers. The SW system also provides cooling to other components, as required, during normal operation. The SW system is not an initiator of any previously evaluated accident. The safety related components associated with SW cooling have been analyzed for a maximum UHS temperature of 92°F. The proposed change maintains this maximum UHS temperature. As such, the qualification of safety related components is not affected. Therefore, the probability of occurrence of a previously evaluated accident is not increased.

The new maximum 24 hour average UHS water temperature limit of 90.5°F has been evaluated and it was determined that the SW system will maintain sufficient heat removal capability. Existing TS operability requirements for the UHS ensure that conservatively bounding assumptions used in the analysis of the SW system's heat removal capability will be met, or the UHS will be declared inoperable. As such, the consequences of previously analyzed accidents are not affected[.]

2. Operation with the maximum 24 hour average UHS water temperature as high as 90.5°F will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Increasing the maximum 24 hour average UHS water temperature does not create the possibility of an accident of a different type than any evaluated previously in the safety analysis report. UHS water temperature does not represent an accident initiator. There is no physical change to any plant structure, system, or components. Therefore, there is no possibility of an accident of a different type.

Increasing the maximum 24 hour average UHS water temperature does not create the possibility of a malfunction of a different type than any evaluated previously. The safety related components associated with SW cooling have been analyzed for a maximum UHS temperature of 92°F. This maximum UHS temperature is maintained by the proposed change. As such, this condition does not introduce the possibility of a malfunction of a different type than any evaluated.

3. Operation with the maximum 24 hour average UHS water temperature as high as 90.5°F does not involve a significant reduction in a margin of safety.

UHS temperature limits are established to ensure that the SW system is able to provide sufficient cooling water for the removal of heat from equipment, such as the DGs, RHR pump seal coolers, ECCS room cooling units, and RHRSW heat exchangers, required for a safe reactor shutdown following a DBA or transient. CP&L has performed an analysis which demonstrates that this capability is not reduced with the increased maximum 24 hour average UHS water temperature limit. Existing TS operability requirements for the UHS ensure that conservatively bounding assumptions used in the analysis of the SW system's heat removal capability will be met, or the UHS will be declared inoperable. As such, the ability of the SW system to perform its intended safety function is not affected and the margin of safety is not reduced.

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

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: Richard P. Correia.

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 amendment request: February 15, 2001.

Description of amendment request: The proposed amendment revises Technical Specifications (TS) 3/4.3.2 ''Engineered Safety Features Actuation System Instrumentation,'' 3/4.3.3.1 ''Radiation Monitoring Instrumentation,'' 3/4.6.1.1 ''Containment Integrity,'' 3/4.6.1.7 ''Containment Ventilation System,'' 3/4.6.3 ''Containment Isolation Valves,'' 3/4.9.4 ''Containment Building Penetrations,'' 3/4.9.9 ''Containment Ventilation System Isolation System,'' and associated Bases to clarify and relocate requirements by implementing the guidance of pre-approved NUREG–1431, Revision 1.

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

The proposed changes modify required Actions and Surveillance Requirements previously reviewed and approved by the NRC in improved Technical Specifications (ITS) and changes to ITS as described in TSTF [Technical Specification Traveler Form]–30, TSTF–45, TSTF–46, and TSTF–269. These changes are administrative in nature in that they do not modify the design or operation of Structures, Systems, and Components (SSCs) that initiate or mitigate the consequences of an accident.

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

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

The proposed changes do not involve new plant components or procedures, but only revise existing Technical Specification Actions and Surveillance Requirements. These changes do not modify the design or operation of Structures, Systems, and Components (SSCs) that could initiate an accident.

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

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

The proposed changes modify required Actions and Surveillance Requirements previously reviewed and approved by the NRC in improved Technical Specifications (ITS) and changes to ITS as described in TSTF–30, TSTF–45, TSTF–46, and TSTF–269.

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

NRC Section Chief: Richard P. Correia.

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: February 28, 2001.

Description of amendment request: The proposed amendments would revise the Technical Specifications to incorporate new requirements for the Low Pressure Service Water system standby pump auto start circuitry, related surveillance requirements, and Bases.

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 Low Pressure Service Water (LPSW) Auto-start circuitry provides a means of automatic response to start the standby LPSW pump after the running LPSW pump fails to restart following a Loss Of Offsite Power (LOOP) event.

Loss Of Coolant Accidents (LOCA) events actuate the LPSW pumps via the Engineered Safeguards Systems. This modification will not change this response.

The LPSW pumps automatically restart following a LOOP event. A failure of a running LPSW pump to restart and LPSW header pressure not returning to normal operating values following a LOOP event will actuate the LPSW Standby Pump Auto-Start circuitry. The circuitry will start the LPSW standby pump. When LPSW header pressure returns to normal operating values, the auto-start signal will be cleared from the LPSW pumps start circuits.

The modification enhances plant design basis functions by ensuring that the standby LPSW pump starts to provide flow. This removes the necessity to rely on alternative systems and/or components to mitigate design basis events. It will eliminate a degraded/non-conforming condition, and will support returning affected systems to Maintenance Rule (MR) a(2) status.

This modification does not involve an 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 kind of accident previously evaluated:

No. This modification adds LPSW Standby Pump Auto-Start circuitry such that if the LPSW pumps fail to restart following a LOOP, the standby LPSW pump will start to provide system flow. This enhances current plant design. It ensures system flow and eliminates reliance on alternative systems and/or components that may or may not be safety related to mitigate the design basis event.

This modification will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

No. The proposed change does not adversely affect any plant safety limits, set points, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. The change will enhance the ability to provide flow from the standby LPSW pump following a LOOP. It eliminates reliance on alternative systems and/or components to mitigate the design basis event should the LPSW pumps fail to restart. Therefore, the proposed change does not involve a reduction in a margin of safety.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

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 20005.

NRC Section Chief: Maitri Banerjee, Acting.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: October 30, 2000.

Description of amendment request: Energy Northwest is requesting a revision to the Columbia Generating Station Final Safety Analysis Report (FSAR) in regards to the spent fuel storage and spent fuel cask handling descriptions. There are significant physical differences between the General Electric cask analyzed in the FSAR and the new Holtec HI-STORM 100 cask system. The physical description of the Columbia Generating Station spent fuel pool as discussed in the FSAR, does not accurately reflect the existing configuration. The specific changes to the FSAR include:

1. The FSAR describes two separate pools for spent fuel handling, when there is only one pool. The FSAR states that there is a spent fuel cask storage and a cask loading pool adjacent to the spent fuel pool. There is not a separate spent fuel cask storage and loading pool. There is a spent fuel cask loading pit located within the spent fuel pool. The proposed change is to eliminate references to separate pools and to add a statement that, ''Sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool.''

2. The FSAR states that limitations on reactor building crane travel preclude transporting the spent fuel casks over the spent fuel pool. There are no interlocks that prevent crane movement over the spent fuel cask pit loading area, which is part of the spent fuel pool. There are interlocks that prevent movement over the spent fuel racks. The proposed change is to add the statement to the FSAR that, ''Interlocks on the reactor building crane prevent travel over the spent fuel racks.''

3. The FSAR states that at no time while being transported does the fuel cask pass over any safety related equipment. The cask does pass over a safety-related conduit associated with a fuel pool cooling level instrumentation. The proposed change is to add the statement to the FSAR that, ''At no time while being transported does the cask pass over any safe shutdown equipment.''

4. The FSAR discusses cask loading, handling, and features of construction associated with the GE IF–300 spent fuel cask rather than the Holtec HI–STORM 100 spent fuel cask system, which is the cask system that will be used. The proposed change would accurately describe the HOLTEC HI–STORM 100 spent fuel cask 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 change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences.

Accidents previously evaluated in the FSAR that could be influenced by these FSAR text changes regarding cask handling and spent fuel loading operations include the Spent Fuel Cask Drop Accident (FSAR 15.7.5) and the Fuel Handling Accident (FSAR 15.7.4).

Spent Fuel Cask Drop Accident: Sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool. (Reference: Columbia Generating Station FSAR Section 15.7.5, ''Spent Fuel Cask Drop Accident''). The drop accident is not deemed credible and the revision of the FSAR description will continue to maintain the drop accident as incredible. Additionally, as a defense-in-depth measure, crane position interlocks prevent lifting a spent fuel cask over the spent fuel stored in the pool.

As the cask is moved in and out of the fuel pool, it passes over several cables and conduits supporting plant equipment. They include nonsafety-related cables such as those supplying the refueling bridge, and spent fuel pool temperature indicator FPC–TE–7. Additionally, a safety-related conduit for FPC–LE–5 is included in the cask load path. While a cask drop, which could damage or cut the cable to FPC–LE–5 is not credible, operator error in which the cable is damaged by the cask not clearing the conduit during cask movement may be credible. If the cable were damaged, it might inhibit one train of the automatic isolation signal for the fuel pool cooling system. The automatic isolation of interest occurs on low fuel pool water level, isolating the Seismic Category I cooling portion of the system from the Seismic Category II cleanup portion of the system. A fuel pool low water level coincident with a crane operator damaging the cable for FPC–LE–5 is an extremely low probability event. However, in the case of a damaged cable for FPC–LE–5, automatic isolation on low water level would still occur because a separate, redundant, logic train (from FPC–LE–4) would not be affected and would still be capable of accomplishing the isolation function described in FSAR Section 9.1.3.2.3. The cable for the redundant logic train is not in the cask load path. The cable for FPC–LE–5 also carries a signal for high/low spent fuel pool water level alarm, which has a redundant analogue signal (undamaged in this scenario) from FPC–LS–4.

Fuel Handling Accident: The fuel handling accident is analyzed in FSAR Section 15.7.4. In it, the assumption is made that a failure occurs in a fuel assembly lifting mechanism. The accident which produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle into the reactor core when the reactor pressure vessel (RPV) head is off. The analysis assumes the accident occurs at the maximum height allowed by the fuel handling equipment above spent fuel (34 ft.). Since the same fuel handling mechanism is used in both the reactor (the analyzed accident location) and in the fuel pool, but at a considerably lower available drop height (approximately 3 ft.), the energy available to damage fuel rods is significantly less. As a result, the analyzed fuel handling accident consequences bound the consequences of a fuel assembly drop in the spent fuel pool. Because fuel loaded in a cask will be within approximately 1 ft. [foot] of the elevation of a fuel pool rack, fuel handling for cask loading is essentially the same as other fuel handling within the pool and is also bounded by the FSAR analysis. Therefore the consequences of this accident evaluated previously in the FSAR will not be increased by the proposed change.

The proposed change does not entail any physical alteration to the present plant configuration. Therefore, individual precursors of an accident are unaffected and the probability of an accident previously evaluated is not expected to increase. In addition, since the functions and capabilities of systems designed to operate safely and/or mitigate the consequences of an accident have not changed, the consequences of an accident previously evaluated are not expected to increase.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration.

Information presented in the FSAR describing the spent fuel cask safe load path is revised by this amendment. To agree with the current plant configuration noted above, the FSAR will need to be changed to read, ''At no time while being transported does the cask pass over any safe shutdown equipment.'' The objectives referenced in RG [Regulatory Guide] 1.13, Rev. 1, and the guidelines of NUREG–0612 (to prevent impact by heavy loads with safe shutdown equipment) will continue to be met. The proposed change does not entail any physical alteration to the present plant configuration. There are no new precursors of an accident created and no new or different kinds of accidents are created.

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

There are no plant modifications required as a result of the proposed FSAR change. The proposed FSAR text changes correct inaccuracies partly resulting from incorrect original process descriptions. Since then, there have been significant changes to spent fuel cask handling and design requirements including the necessity for extended dry storage of spent fuel at independent spent fuel storage installations. With the proposed FSAR text changes incorporated, the FSAR will accurately describe actual plant configuration and processes related to spent fuel cask handling and the NRC certified Holtec HI-STORM 100 System.

The Columbia Generating Station reactor building crane is single-failure-proof and therefore no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool. A single-failure-proof crane obviates the need for an isolated spent fuel cask transfer pool. In addition, safe load paths are defined that keep the spent fuel cask away from irradiated fuel and safe shutdown equipment. This is in accordance with defense-in-depth approach as described in NUREG–0612, Section 5.2, ''Bases for Guidelines''.

The proposed FSAR change contains information about Columbia Generating Station spent fuel cask handling that has not been previously reviewed and approved by the NRC; however, there is no safety significance to this FSAR amendment request. The FSAR text corrections are in agreement with applicable regulations and no physical alteration to the plant configuration is required.

Therefore, this 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005–3502.

NRC Section Chief: Stephen Dembek.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: February 20, 2001.

Description of amendment request: The proposed amendment revises the Columbia Generating Station Technical Specifications (TS) to remove selected operating mode restrictions for performing emergency diesel generator (DG) testing. This change will allow the DG testing to be performed during any plant operating mode. The proposed change removes the restriction associated with the following surveillance requirements (SRs) that prohibit performing the required DG testing during Modes 1 and 2.

1. SR 3.8.1.9: This SR requires demonstrating that the DG can reject its single largest load without the DG output frequency exceeding a specific limit.

2. SR 3.8.1.10: This SR requires demonstrating that the DG can reject its full load without the DG output voltage exceeding a specific limit.

3. SR 3.8.1.14: This SR requires starting and then running the DG continuously at or near full-load capability for greater than or equal to 24 hours.

The proposed change also removes the restriction associated with the following SRs that prohibits performing the required testing during Modes 1, 2, and 3.

1. SR 3.8.1.13: This SR requires demonstrating that the DG non-emergency (non-critical) automatic trips are bypassed on an actual or simulated emergency core cooling system (ECCS) initiation signal.

2. SR 3.8.1.17: This SR requires demonstrating that the DG automatic switchover from the test mode to ready-to-load operation is attained upon receipt of an ECCS initiation signal while maintaining availability of the offsite source.

The proposed change also allows the performance of SR 3.8.1.14 to satisfy SR 3.8.1.3 (monthly one-hour synchronized and loaded DG run) by adding a Note 5 to SR 3.8.1.3 that allows the endurance and margin test of SR 3.8.1.14 to be performed in lieu of load-run test in SR 3.8.1.3.

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 DGs and their associated emergency loads are accident mitigating features, not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes. As such, the ability of the DGs to respond to a design basis accident will not be adversely impacted by these proposed changes. The proposed changes do not result in a plant configuration change for performance of the additional testing different from that currently allowed by the Technical Specifications. In addition, experience and further evaluation of the probability of a DG being rendered inoperable concurrent with or due to a significant grid disturbance support the conclusion that the proposed changes do not involve any significant increase in the likelihood of a loss of safety bus. Therefore, there would be no significant impact on any accident consequences.

Based on the above, the proposed change to permit certain DG surveillance tests to be performed during plant operation will not involve a significant increase on accident probabilities or consequences.

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

No new accident causal mechanisms would be created as a result of NRC approval of this amendment request since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration currently allowed by other DG SRs that currently allow testing in plant Modes 1, 2 and 3. An interaction between the DG under test and the offsite power system that could lead to a consequential loss of safety bus during a grid disturbance is not deemed to be credible. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

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

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

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the plant DGs do not affect the operability requirements for the DGs, as verification of such operability will continue to be performed as required (except during different allowed Modes). Continued verification of operability supports the capability of the DGs to perform their required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment.

In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted. Therefore, implementation of the proposed changes would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: January 24, 2001.

Description of amendment request: The license amendment request consists of changes to the Technical Specifications (TSs) to revise the reactor vessel pressure/temperature (P/T or P–T) limits specified in TS 3.4.11, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits,'' for reactor heatup, cooldown, and critical operation, as well as for inservice leak and hydraulic tests for the RCS. Also, the current RCS P/T Limits in TS Figure 3.4–11, ''Minimum Temperature Required Vs. RCS Pressure,'' would be replaced with recalculated RCS P/T limits based, in part, on an alternate methodology. The alternate methodology uses American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Code) Case N–640, ''Alternative Requirement Fracture Toughness for Development of P–T Limit Curves for ASME B&PV Code Section XI, Division 1,'' for alternate reference fracture toughness for reactor vessel materials in determining the P/T limits.

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

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

The proposed changes to the River Bend [Station] reactor coolant system (RCS) pressure/temperature (P/T) limits do not modify the boundary, operating pressure, materials or seismic loading of the reactor coolant system. The proposed changes do adjust the P/T limits for radiation effects to ensure that the RPV [reactor pressure vessel] fracture toughness is consistent with analysis assumptions and NRC [Nuclear Regulatory Commission] regulations. An evaluation has been performed justifying the use of the methodology contained in Code Case N–640 to determine the P–T curve. The proposed P/T limits were determined using this methodology. Thus, the proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated. The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected.

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

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

The proposed changes to the reactor pressure vessel pressure-temperature limits do not affect the assumed accident performance of any structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

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

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

The methodology for determining the RCS P/T limits ensures that the limits provide a margin of safety to the conditions at which brittle fracture may occur. The methodology is based on requirements set forth in Appendix G and Appendix H of 10 CFR [Part] 50, with reference to the requirements and guidance of ASME Section XI, and on guidance provided in Regulatory Guide 1.99, Revision 2. The revised P/T limits are also based on this methodology except as modified by application of the noted Code Case. Although the Code Case constitutes relaxation from the current requirements of 10 CFR [Part] 50 Appendix G, the alternatives allowed by the Code are based on industry experience gained since the inception of the 10 CFR [Part] 50 Appendix G requirements for which some of the requirements have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code Case maintain a margin of safety that is consistent with the intent of 10 CFR [Part] 50 Appendix G, i.e., with regard to the margin originally contemplated by 10 CFR

[Part] 50 Appendix G for determination of RPV/RCS P/T limits.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 24, 2001.

Description of amendment request: The amendment request proposes that the River Bend Station Operating License be amended to change the limit on the Low Power Setpoint Limit specified by Technical Specifications 3.1.3 ''Control Rod OPERABILITY,'' 3.1.6 ''Control Rod Pattern,'' and 3.3.2.1 ''Control Rod Block Instrumentation'' from less than or equal to 20% reactor thermal power to less than or equal to 10% reactor thermal power.

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

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change revises the setpoint from 20% to 10% rated power and does not affect the function, reliability or required surveillance frequency of the RPC [Rod Pattern Control] set forth in the Technical Specification. It does not constitute a safety significant change to the plant design or operation since the RPC and associated BPWS [Banked Position Withdrwawal Sequence] will continue to ensure site compliance with 10 CFR [Code of Federal Regulations Part] 100.

The RPC limits the incremental worth of control rods during reactor startup and shutdown. The BPWS allows continuous withdrawal from fully inserted to the fully withdrawn position for the first 25% of control rod density. The change in LPSP [Low Power Setpoint Limit] does not affect any of the parameters or conditions that contribute to initiation of the control rod drop accident since it is not the precursor of the accident. On this basis, change in the low power setpoint will not increase the probability of an accident previously evaluated.

The low power setpoint of the RPC is set so that the resultant peak fuel enthalpy due to the postulated rod drop accident shall be equal to or less than 280 cal/gm. For operation below the LPSP, systems are provided so that the design limit of 280 cal/gm is not exceeded for the design basis accident. Conformance to the 280 cal/gm design limit also ensures that the 10 CFR [Part] 100 offsite dose criteria will not be exceeded for the design basis accident. GE [General Electric] generic analysis demonstrates the radiological effect following a CRDA [Control Rod Drop Accident], for all current GE fuel design is within the guidelines set forth in 10 CFR [Part] 100. No River Bend specific analysis is necessary. On these bases, the proposed LPSP reduction does not significantly change the consequences of an accident previously evaluated.

2. The request does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

The LPSP is set so that the resultant peak fuel enthalpy due to the postulated rod drop accident at power levels below the LPSP, shall be equal to or less than 280 cal/gm, ensuring compliance with 10 CFR [Part] 100 offsite dose criteria. The proposed change implements the reduction in LPSP from 20% to 10% of rated power without the addition of new hardware.

The change in LPSP does not affect any of the parameters or conditions that contribute to initiation of any accident since the LPSP is not the precursor of any accident. The LPSP is the point at which the RPCS [Rod Pattern Control System] switches between the RPC and RWL [Rod Withdrawal Limit] function. Periodic verification that it is within the allowable value is required. The proposed change does not affect the function and the reliability of the RPC, or the required surveillance frequency of Technical Specification LCO [Limiting Condition for Operation]. Furthermore, the reduction in setpoint can be implemented without the addition of new hardware. On this basis, reduction in the low power setpoint does not create the possibility of occurrence of a new or different accident.

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

Below the LPSP, mitigating systems and procedures are used to limit the consequences of a postulated CRDA. These involve a time consuming process of a series of controlled rod moves or steps. The setpoint change has the potential to impact the margin of safety and as such, a series of evaluations and under the worst case scenario were performed for a CRDA. NEDO–10527 demonstrates that a CRDA at or above 10% of rated power will always result in peak fuel enthalpies less than 280 cal/gm. These results assumed the worst single operator error, conservative Technical Specification scram times and rod drop velocity. This generic analysis also included the effect of core and fuel cycle design parameters such as the axial gadolinia distributions. The results indicate, that even for this worst case scenario, the resultant peak fuel enthalpy will always be less than 280 cal/gm, ensuring conformance with guidelines set forth in 10 CFR [Part] 100. Additional vendor analyses show that ''Above approximately 10% power, the RDA cannot exceed 280 cal/gm because of the prompt Doppler feedback in the power range and the impossibility of achieving high rod reactivity worth with the relatively low rod density, even with erroneous rod patterns.'' Finally, the new models, which include moderator reactivity feedback, provide additional justification for the 10% of rated power LPSP. These methods indicate that the existence of any steam flow (i.e., power) will result in the CRDA results remaining below the design basis limit. Therefore, a LPSP limit of 10% is conservative relative to the new models. On these bases, the proposed reduction in the LPSP does not change the margin of safety significantly.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 24, 2001.

Description of amendment request: The request consists of a change to Technical Specification 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' to permit the operation of the Inclined Fuel Transfer System (IFTS) bottom valve after removal of the IFTS primary containment isolation blind flange while the containment is required to be operable.

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 significantly increase the probability or consequences of an accident previously evaluated.

The proposed change permits the operation of the IFTS Bottom valve after removal of the inclined fuel transfer system (IFTS) primary containment isolation blind flange when primary containment operability is required in MODE 1, 2, and 3. This will permit the full operation of the IFTS while the plant is operating. With respect to the probability of an accident, this aspect of the containment structure does not directly interface with the reactor coolant pressure boundary. Operation of the IFTS bottom valve after the removal of the blind flange does not involve modifications to plant systems or design parameters that could contribute to the initiation of any accidents previously evaluated. Operation of IFTS is unrelated to the operation of the reactor, and there is no aspect of IFTS operation that could lead to or contribute to the probability of occurrence of an accident previously evaluated. Operation of the IFTS bottom valve during operation of IFTS system after removal of the blind flange does not result in changes to procedures that could impact the occurrence of an accident.

With respect to the issue of consequences of an accident, the function of the containment is to mitigate the radiological consequences of a loss of coolant accident (LOCA) or other postulated events that could result in radiation being released from the fuel inside containment. While the proposed change does not change the plant design, it does permit an alteration of the containment boundary for the IFTS penetration. Altering the containment boundary in this case (i.e., Opening the IFTS bottom valve) would not result in any additional IFTS components being subjected to containment pressure in the event of a LOCA. However, the additional post-accident peak pressure load to be imposed upon the components in the IFTS if the blind flange is removed is a small fraction of their design capability. Therefore, they are considered an acceptable barrier to prevent uncontrolled release of post-accident fission products for this proposed change.

As discussed in LAR [License Amendment Request] 1999–30, the proposed change required examination of two potential leakage pathways. The larger is the IFTS transfer tube, itself. The other, much smaller one, is a branch line used for draining the IFTS transfer tube during its operation. The bottom of the IFTS transfer tube is always water sealed, and maintained so by the submergence of the water in the transfer tube and in the fuel building spent fuel storage pool (the lower pool). The height of this water seal is greater than that necessary to prevent leakage from the bottom of the transfer tube during accidents that result in the calculated peak post-DBA [design basis accident] LOCA pressure, P_a. The potential leakage pathway from the drain piping that attaches to the transfer tube will be isolated if required, via administrative controls on the drain piping isolation valve. Additionally, as committed to in LAR 1999–30, the drain piping isolation valve will be added to the Primary Containment Leakage Rate Testing Program (Technical Specification 5.5.13) to ensure that leakage past this valve will be maintained consistent with the leakage rate assumptions of the accident analysis. Due to the test methodology, the portion of the large transfer tube piping outboard of the blind flange (the portion of the tube which becomes exposed to the containment atmosphere during the draining portion of the IFTS operation) will also be part of the leakage rate test boundary and will therefore also be tested. Therefore, no unidentified leakage will exist from the piping and components that are outboard of the blind flange, and the leakage rate assumptions of the accident analysis will be maintained.

Therefore, the proposed change does not result in a significant increase in the probability or the consequences of previously evaluated accidents.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous analyzed.

The proposed change consists of permitting operation of the IFTS Bottom valve after the removal of a the IFTS Blind Flange which is not part of the primary reactor coolant pressure boundary nor involved in the operation or shutdown of the reactor. Being passive, the presence or absence of the IFTS Blind Flanges does not affect any of the parameters or conditions that could contribute to the initiation of any incidents or accidents that are created from a loss of coolant or an insertion of positive reactivity. Realigning the boundary of the primary containment to include portions of the IFTS is also passive in nature and therefore has no influence on, nor does it contribute to the possibility of a new or different kind of incident, accident or malfunction from those previously analyzed. Furthermore, operation of the IFTS is unrelated to the operation of the reactor and there is no mishap in the process that can lead to or contribute to the possibility of losing any coolant from the reactor or introducing the chance for an insertion of positive or negative reactivity, or any other accidents different from and not bounded by those previously evaluated.

Therefore, the proposed change does not result in creating the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change involves the operation of the IFTS Bottom Valve after realignment of the primary containment boundary by removing the blind flange which is a passive component. The margin of safety that has the potential of being impacted by the proposed change involves the dose consequences of postulated accidents which are directly related to potential leakage through the primary containment boundary. The potential leakage pathways due to the proposed change have been reviewed, and leakage can only occur from the administratively controlled IFTS transfer tube drain piping, and from the IFTS transfer tube itself. A dedicated individual will be designated to provide timely isolation of this drain piping during the duration of time when this proposed change is in effect. The conservatively calculated dose which might be received by the designated individual while isolating the drain piping is calculated to be 3.8 rem [roentgen equivalent man] TEDE [Total Effective Dose Equivalent], which remains within the guidelines of General Design Criterion (GDC) 19 (10 CFR [Code of Federal Regulations Part] 50, Appendix A, Criterion 19). Furthermore, the drain piping isolation valve will be added to the Primary Containment Leakage Rate Testing Program (Technical Specification 5.5.13) to ensure that leakage from the piping and components located outboard of the blind flange will be maintained consistent with the leakage rate assumptions of the accident analysis.

Studies of the capability of the IFTS system to withstand containment pressurization under severe accident conditions have been conducted. These studies conclude that IFTS, including the transfer tube and its valves, has a capability to withstand beyond design basis severe accident containment pressures which is greater than that of the containment structure itself. The RBS [River Bend Station] Emergency Operating Procedures (EOPs) are based on an ultimate containment failure pressure capability of 53 psig [pounds per square inch gauge], which represents a margin of safety of 38 psi [pounds per square inch] above the 15 psig containment design pressure.

This capability to withstand containment pressurization under severe accident conditions envelops other non-DBA LOCA scenarios, such as the small break LOCA. For the large break LOCA, additional defense-in-depth is provided by maintaining a water seal greater than P_a above the outlet of the IFTS transfer tube in the lower pool.

The RBS base LERF [Large Early Release Frequency] is 5.915E–9/yr. Removal of the blind flange increases the LERF by 6.315E–9/yr to 1.223E–8/yr. This increase in LERF is due to the reduced failure pressure of the IFTS tube. With the blind flange installed, the IFTS tube has a median failure pressure of approximately 80 psig. The IFTS tube was evaluated to withstand a pressure of 40 psig, with the blind flange removed. This lower IFTS failure pressure increases the probability of gross failure versus penetration failure at a given containment pressure. This shift in failure probability means that some of the less severe pressurization events (i.e. small hydrogen deflagrations) have a higher probability of causing a LERF. Based on the RBS PRA [Probabilistic Risk Assessment] Analysis, the operation of the bottom valve has no affect on LERF.

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

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: February 14, 2001.

Description of amendment request: The amendment would modify the

Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs) to extend the allowed outage time (AOT) for the emergency diesel generators (EDGs) and the associated fuel oil storage tanks (FOSTs) from 72 hours to 14 days on a one-time basis.

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 evaluated?

No. The proposed License amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The EDGs and their associated fuel oil systems are not part of any accident initiation; therefore there is no increase in the probability of an accident.

At a minimum, two EDGs are still available with sufficient fuel oil supply to mitigate IP3 design basis accidents. The minimum safeguards equipment can still be powered even if one EDG and FOST is assumed to be lost due to single failure. This has been verified by EDG loading calculation, IP3–CALC–ED–00207, ''480V Bus 2A, 3A, 5A & 6A and EDGs 31, 32 and 33 Accident Loading''. With the associated EDG available and aligned for automatic start capability (although declared inoperable) during this EDG FOST outage, further backup to the remaining two EDGs is provided. By the design of the overall EDG fuel oil system, the associated EDG fuel oil day tank is able to be supplied with sufficient fuel oil supply from either of the remaining two FOSTs, via their transfer pumps, in order to support operation of this associated EDG, if necessary.

To support fuel oil needs of all three EDGs, if necessary, the FSAR [final safety analysis report] describes that additional fuel oil supplies are available on the Indian Point site and locally near the site. Further EDG fuel oil supplies are available in the region, about 40 miles from IP3. Overall, the EDGs are designed as backup AC power sources in the event of a Loss of Offsite Power (LOOP). The proposed one-time AOT for each EDG/FOST does not change the conditions or minimum amount of safeguards equipment assumed in the safety analysis for design basis accident mitigation, since a minimum of two EDGs is assumed. No changes are proposed as to how the EDGs provide plant protection. Additionally, no new modes of overall plant operation are proposed as a result of this change. A PRA [probablistic risk assessment] evaluation determined that the conditional core damage probability (CCDP) for these scenarios is less than the threshold value of 1 E–6. Therefore, the proposed one-time license amendment to TS 3.7.B.1 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed License amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not introduce any new overall modes of plant operation or make any permanent physical changes to plant systems necessary for effective accident mitigation. The minimum required EDG operation remains unchanged by removal of a single FOST for repair. Additionally, added requirements to minimize risk associated with loss of offsite power also support this one-time extended AOT. Also, as previously stated, the EDGs and FOSTs are not part of any accident initiation scenario. Therefore the proposed one-time license amendment to TS 3.7.B.1 does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Does the proposed License amendment involve a significant reduction in a margin of safety?

No. The proposed License amendment does not involve a significant reduction in a margin of safety. The minimum safeguards loads can be maintained available if needed for design basis accident mitigation with two EDGs operable combined with their respective FOSTs. The selected, inoperable EDG will be available and aligned for automatic start capability (though declared inoperable) during this outage. The additional fuel oil needed to support three EDGs in this condition is available as indicated in the present design and licensing basis. The FSAR describes that this fuel can be provided from the Indian Point site, local sources and from a source about 40 miles away to support the additional 30,026 gallons TS required fuel oil, already existing at the Buchanan substation. Therefore, sufficient fuel oil will be available for potential events that could occur during this 14-day AOT. The PRA evaluation for the case of maintaining the 31, 32 or 33 EDG available (though declared inoperable) with its FOST out for repair indicates an acceptable safety margin below the risk-informed threshold of 1 E–6.

The 480VAC electrical distribution system can be fed from a number of TS independent 13.8kV and 138kV offsite power sources to minimize reliance of IP3 on EDG power sources during the extended AOT requested. Additional requirements to minimize risk associated with the potential for loss of offsite power sources within this TS change also ensure that this extended AOT does not involve a significant reduction in safety margin. On this basis, the proposed one-time license amendment to TS 3.7.B.1 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: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Generating Station, 600 Rocky Hill Road, Plymouth, MA 02360.

NRC Section Chief: Marsha Gamberoni.

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 amendment request: January 25, 2001.

Description of amendment request: Entergy Operations, Inc. is proposing that the Grand Gulf Nuclear Station (GGNS) Operating License be amended to revise the GGNS Technical Specification (TS), Surveillance Requirement (SR) 3.1.4.2 to increase the control rod scram time testing interval from 120 days to 200 days of full power operation. The licensee also proposes to revise the associated TS Bases to reflect the proposed revision to the SR.

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

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

The proposed change will not adversely impact plant operation. There will be no change in the method of performing the tests. The extended test frequency will provide some positive safety benefits by reducing the complexity of half of the control rod sequence exchange maneuvers, reducing the likelihood of a reactivity or fuel related event.

The actual rod insertion times and control rod reliability are not impacted by this proposed change; only the probability of detecting slow rods is impacted. The potential consequence of the proposed change is that one or more slow rods that would have been detected under the current 120-day frequency, may not be detected due to a reduced number of tests under the 200-day frequency.

Historical data shows that the GGNS control rod insertion function is highly reliable and rod insertion tests meet the scram time limits 99.84% of the time. Statistical analysis also demonstrates that the extended frequency would have little impact on the ability to detect slow rods in the sampling tests.

There is no safety consequence resulting from ''slow'' rods so long as the plant does not exceed the Technical Specification 3.1.4 Limiting Condition of Operation [LCO] requirement of no more than 14 slow rods in the entire core or no two OPERABLE ''slow'' rods occupying adjacent positions. It is highly unlikely that a combination of missed detections and known ''slow'' rods would lead to the requirement to take action in accordance with TS 3.1.4. Therefore, it is highly unlikely that the reduction in test frequency would have any impact on plant operation or safety.

The analysis assumes that all 14 slow rods take 7 seconds to reach notch position 13 which is very conservative base on actual rod performance. Control rod data shows that rods that have failed the time requirements are usually only a fraction of a second slower. In the unlikely event that, due to the reduction of test frequency, the plant is unknowingly operating with one or two more slow rods than the 14 slow control rods permitted by the LCO, the consequences would still be insignificant. The low probability of MODE 1 operation with excess slow rods combined with the low consequence of a few excess slow rods, leads to the conclusion that the probability or consequences of accidents previously evaluated are not significantly increased.

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

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

The proposed change will make no change to plant configuration or test procedures. The proposed change does not impact the operation of the plant except to reduce the number of required tests and slightly increase the probability of failing to detect a slow control rod. Operating with possibly one or two undetected slow rods does not create the possibility of an accident, since sudden control rod insertion by scram only occurs during the mitigation of accidents.

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

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

The GGNS accident analyses assume a certain negative reactivity time function associated with scrams. So long as the LCO of Technical Specification 3.1.4 is met, that is, there are no more than 14 slow control rods in the entire core or two OPERABLE ''slow'' rods occupying adjacent locations, all accident analysis assumptions are met and there is no reduction in any margin of safety. The proposed change does not impact the Technical Specification LCO, or any other allowable operating condition. The potential for an increase in the probability of being outside acceptable operating conditions due to this proposed change is insignificant. Calculations have demonstrated that the likelihood of detecting four slow rods with proposed testing frequency over a fuel cycle is lower than that with the current testing frequency by a negligible amount (2E–O7). The difference is even smaller for detecting greater number of slow rods over a cycle. Therefore, since there is no impact on allowable operating parameters and the likelihood of detecting significant numbers of slow rods is only negligibly affected, there is no significant reduction in a margin of safety.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

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

Date of amendment request: November 30, 2000.

Description of amendment request: The proposed amendment would revise the ''Diesel Fuel Oil Testing Program'' in technical specifications to relocate the specific American Society for Testing Materials (ASTM) standard reference from the Administrative Controls Section of TS to a licensee-controlled document, i.e., the Diesel Fuel Oil Program in the Technical Requirements Manual (TRM). In addition, the ''clear and bright'' test has been expanded to allow a water and sediment content test to establish the acceptability of new fuel oil. The proposed changes are consistent with changes previously approved by the Nuclear Regulatory Commission (NRC).

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

The proposed changes relocate the specific diesel fuel oil related American Society for Testing and Materials (ASTM) Standard reference from the Administrative Controls Section of Technical Specifications (TS) to a licensee-controlled document, i.e., the Diesel Fuel Oil Program in the Technical Requirements Manual (TRM). The Braidwood Station and the Byron Station TRM is incorporated by reference in the Braidwood and Byron Stations' Updated Final Safety Analysis Report (UFSAR). Since any change to these licensee-controlled documents will be evaluated pursuant to the requirements of 10 CFR 50.59, ''Changes, tests and experiments,'' no increase in the probability or consequences of an accident previously evaluated is involved. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil in lieu of the ''clear and bright'' test. We consider that the quantitative water and sediment test is equivalent to the qualitative clear and bright test.

Relocating the specific ASTM Standard references from the TS to a licensee-controlled document (i.e., the Diesel Fuel Oil Program in the TRM), and allowing a water and sediment content test to be performed to establish the acceptability of new fuel oil, will not affect nor degrade the ability of the safety-related diesel generators (DGs) (i.e., the Emergency DG and the Auxiliary Feedwater pump DG) to perform their specified safety function. Fuel oil quality will continue to meet ASTM requirements.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Braidwood and Byron Stations' UFSAR. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Braidwood and Byron Stations' UFSAR.

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

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

The proposed changes relocate the specific ASTM Standard reference from the Administrative Controls Section of TS to a licensee-controlled document, i.e., the Diesel Fuel Oil Program in the TRM. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil.

The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes relocate the specific ASTM Standard reference from the Administrative Controls Section of TS to a licensee-controlled document, i.e., the Diesel Fuel Oil Program in the TRM. Instituting the proposed changes will continue to ensure the use of current applicable ASTM Standards to evaluate the quality of both new and stored fuel oil designated for use in the safety-related DGs. The detail associated with the specific ASTM Standard reference is not required to be in the TS to provide adequate protection of the public health and safety, since the TS still retain the requirement for compliance with the applicable ASTM Standard. Changes to the TRM are evaluated in accordance with 10 CFR 50.59. Should it be determined that future changes involve a potential reduction in a margin of safety, NRC review and approval would be necessary prior to implementation of the changes. This approach provides an effective level of control and provides for a more appropriate change control process. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil in lieu of the ''clear and bright'' test. The level of safety of facility operation is unaffected by the proposed changes since there is no change to the TS requirements intended to assure that fuel oil is of the appropriate quality for safety-related DG use. The proposed changes provide the flexibility needed to maintain state-of-the-art technology in fuel oil sampling and analysis methodology.

Therefore, the 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: February 20, 2001.

Description of amendment request: The proposed amendments would increase the allowed outage time from 3 days to 14 days for a single inoperable Division 1 or 2 emergency diesel generator.

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:

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

The proposed changes include the extension of the completion time for the Emergency Diesel Generators (EDGs) from 72 hours to 14 days to allow on-line preventive maintenance to be performed. The EDGs are not initiators of previously evaluated postulated accidents. Extending the completion times of the EDGs would not have any impact on the frequency of any accident previously evaluated, and therefore the probability of a previously analyzed accident is unchanged. The proposed change to the completion time for EDGs will not result in any changes to the plant activities associated with EDG maintenance, but rather will enable a more efficient planning and scheduling of maintenance activities that will minimize potential adverse interactions with concurrent outage activities.

The consequences of a previously analyzed event are the same during a 72 hour EDG completion time as the consequences during a 14 day completion time. Thus the consequences of accidents previously analyzed are unchanged between the existing TS requirements and the proposed change. In the worst case scenario, the ability to mitigate the consequences of any accident previously analyzed is preserved. The consequences of an accident are independent of the time the EDGs are out-of-service. As a general practice, no other additional failures are postulated while equipment is inoperable within its TS completion time.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed changes do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the change involve a significant reduction in a margin of safety?

The proposed changes will extend the allowable completion times for the Required Actions associated with restoration of an inoperable Division 1 or Division 2 EDG. The proposed 14 day EDG completion time is based upon both a deterministic evaluation and a risk-informed assessment. The availability of offsite power coupled with the availability of the opposite unit EDG via the unit cross-tie breaker and the use of the Configuration Risk Management Program (CRMP) provide adequate compensation for the potential small incremental increase in plant risk of the EDG extended completion time. In addition, the increased availability of the EDGs during refueling outage offsets the small increase in plant risk during operation. The proposed EDG extended completion times in conjunction with the availability of the opposite unit EDG continues to provide adequate assurance of the capability to provide power to the Engineered Safety Feature (ESF) buses. The risk assessment concluded that the increase in plant risk is small and consistent with the NRC's Safety Goal Policy Statement, ''Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,'' **Federal Register**, Volume 60, p. 42622, August 16, 1995, and guidance contained in Regulatory Guides (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,'' dated July, 1998, and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,'' dated August, 1998. Together, the deterministic evaluation and the risk-informed assessment provide high assurance of the capability to provide power to the ESF buses during the proposed 14 day EDG completion time.

Therefore, implementation of the proposed changes 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Anthony J. Mendiola.

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

Date of application for amendment: February 21, 2001 (TS–265).

Brief description of amendment: The proposed amendment would revise the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) 3.3.8 to clarify the actions to be taken in the event that one or more channels of loss of voltage or degraded voltage Emergency Diesel Generator (EDG) start functions become inoperable.

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

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

The emergency diesel generator (EDG) loss of power start is not an initiator of any design basis accident. The EDG loss of power start is intended to protect engineered safeguards equipment from damage due to sustained undervoltage conditions, and to ensure rapid restoration of power to the engineered safeguards electrical buses in the event of a loss of offsite power.

The proposed license amendment clarifies the actions to be taken in the event that one or more channels of the undervoltage or degraded voltage start Functions become inoperable. The design functions of the EDG loss of power start and the initial conditions for accidents that require an EDG loss of power start will not be effected by the change. Therefore, the change will not increase the probability or consequences of an accident previously evaluated.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed amendment involves no changes to the design or operation of the EDG loss of power start. The proposed changes will ensure that the EDGs and engineered safeguards actuation system (ESAS) automatic initiation logic perform as assumed in the safety analysis in the event of a loss of offsite power. The proposed change will not affect other EDG or ESAS functions, and will not create any new plant configurations. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does not involve a significant reduction in the margin of safety.

The proposed amendment clarifies the actions to be taken in the event one or more undervoltage or degraded voltage start Functions become inoperable. The proposed changes ensure appropriate actions are taken to restore the operability of the EDG loss of power start under these conditions. Thus, the proposed amendment will not result in a reduction in the margin of safety.

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

Attorney for licensee: R. Alexander Glenn, Associate General Counsel, Florida Power Corporation, MAC–A5A, P.O. Box 14042, St. Petersburg, Florida, 33733–4042.

NRR Section Chief: Richard P. Correia.

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

Date of amendment request: February 21, 2001 (TS–266).

Description of amendment request: The changes proposed revise various administrative actions, requirements, and responsibilities contained in Improved Technical Specifications (ITS) 2.0, Safety Limits, and ITS 5.0, Administrative Controls, to reflect the recent CR–3 Nuclear Operations re-organization and the amended requirements of 10 CFR 50.72, 10 CFR 50.73 and 10 CFR 50.59.

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 not involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed license amendment deletes redundant administrative requirements contained in ITS 2.0, ''Safety Limits'' and updates position titles in ITS 5.0, ''Administrative Controls,'' to reflect the current CR–3 Nuclear Operations organization. The design functions of the structures, systems and components at CR–3, and the initial conditions for the analyzed accidents at CR–3 will not be affected by the change. Therefore, the change will not increase the probability or consequences of an accident previously evaluated.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed.

The changes proposed by this amendment are administrative in nature. The proposed amendment involves no changes to the design, function or operation of any structure, system or component at CR–3 and will not result in any new plant configurations. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does not involve a significant reduction in the margin of safety.

The proposed changes are administrative in nature. The safety margins established through the design and facility license, including the CR–3 Improved Technical Specifications will not be changed by the proposed amendment. In addition, the proposed changes will ensure that administrative requirements and responsibilities contained in the ITS are consistent with the current CR–3 Nuclear Operations organization as described in the CR–3 Final Safety Analysis Report and the requirements specified in 10 CFR 50.72, 10 CFR 50.73 and 10 CFR 50.59. Thus, the proposed amendment will not result in a reduction in the margin of safety.

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

Attorney for licensee: R. Alexander Glenn, Associate General Counsel (MAC–BT15A), Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard P. Correia.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of amendment request: January 19, 2001.

Description of amendment request: The proposed amendment would extend surveillance intervals associated with the emergency diesel generators and station batteries to preclude a mid-cycle shutdown of the unit.

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 license conditions do not affect or create any accident initiators or precursors. As such, the proposed license conditions do not increase the probability of an accident. The proposed license conditions do not involve operation of the required electrical power sources in a manner or configuration different from those previously recognized or evaluated.

The proposed EDG [emergency diesel generator] engine SR [surveillance requirement] revision involves deferral of the 4.8.1.1.2.e.1 requirement to the next refueling outage and does not reduce the required operable power sources of the Limiting Condition for Operation, does not increase the allowed outage time of any required operable power supplies, and does not reduce the requirement to know that the deferred SRs could be met at all times. Deferral of the testing does not increase by itself the potential that the testing would not be met. The monthly EDG engine starts, fuel level checks, and fuel transfer pump checks will continue to be performed to provide adequate confidence that the required EDG engine will be available if needed. Therefore, it is concluded that the required A.C. sources will remain available and the previously evaluated consequences will not be increased.

The deferral of the battery service tests described above to the refueling outage does not involve any physical changes to the plant or to the manner in which the plant is operated. Therefore, the probability of an accident previously evaluated is not increased. The weekly and quarterly testing, performance monitoring by the system manager, and the current condition of the batteries (e.g., above 100 percent capacity) provide assurance that battery condition and performance will not deteriorate during the deferral period. Therefore, the consequences of the analyzed accidents for CNP [Cook Nuclear Plant] will not be increased due to the deferral of these station battery SRs.

Therefore, based on the above discussion, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed license condition does not involve a physical alteration of the EDG engines or a change to the way the A.C. power system is operated. The proposed license condition does not involve operation of the required electrical power sources in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms of the A.C. power supplies are introduced by extension of the subject SR intervals.

The proposed license conditions for deferral of the station battery SRs listed above to the refueling outage do not involve any physical changes to the plant or to the manner in which the plant D.C. power systems are operated. No new failure mechanisms will be introduced by the SR deferral.

Therefore, the proposed license condition does not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Deferral of the specified EDG engine SR does not introduce by itself a failure mechanism, and past performance of the SR has demonstrated reliability in passing the deferred SRs. The required operable power supplies have not been reduced. Therefore, the availability of power supplies assumed for accident mitigation is not significantly reduced and previous margins of safety are maintained.

The deferral of the station battery SRs to the refueling outage does not involve any physical changes to the plant or to the manner in which the plant is operated. Continuing weekly and quarterly testing, performance monitoring, and the current condition of the batteries provides assurance that the battery condition and performance will be acceptable during the deferral period in that degradations that may occur will be detected. Therefore, the equipment response to accident conditions during the deferral period will not be affected. Thus, the one-time deferral of these 18-month battery service test SRs does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed amendment involves no significant hazards consideration.

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: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: January 3, 2001.

Description of amendment request: The proposed amendment would terminate license jurisdiction for a portion of the Maine Yankee Atomic Power Station site, thereby releasing these lands from Facility Operating License No. DPR–36. The release of these lands will facilitate the donation of this property to an environmental organization pursuant to a Federal Energy Regulatory Commission-approved settlement between Maine Yankee Atomic Power Company and its ratepayers. The lands donated will be used to create a nature preserve and an environmental education center.

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 requested license amendment involves release of land presently considered part of the Maine Yankee plant site under license DPR–36. The land in question is not used for any licensed activities. No radiological materials have historically been used on this land and the land will not be used to support ongoing decommissioning operations and activities.

Most of the land to be released is outside the Exclusion Area Boundary and therefore is not affected by the consequences of any postulated accident. A small portion of the land is within the Exclusion Area Boundary. Maine Yankee will retain sufficient control over activities performed within this land through rights granted in the legal land conveyance documents to ensure that there is no impact on consequences from postulated accidents. Therefore, the release of the land from the Part 50 license will not increase 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 requested amendment involves release of land presently considered part of the Maine Yankee plant site under license DPR–36. The land is not used for any licensed activities or decommissioning operations. The proposed action does not affect plant systems, structures or components in any way. The requested release of the land does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use. This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Additionally, the State of Maine, through legislation, has imposed a 10 mrem/yr all pathways limit, with no more than 4 mrem/yr attributable to drinking water sources. Since the survey results described in Attachments III and IV demonstrate compliance with the radiological criteria for license termination for unrestricted use and demonstrate compliance with the more stringent Maine Standard, therefore, the margin of safety will not be reduced as a result of the proposed release of the nonimpacted land. In fact, since the area is nonimpacted, by definition, there will be no additional dose to the average member of the critical group.

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

Attorney for licensee: Joseph Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578.

NRC Section Chief: Robert A. Gramm.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of amendment request: February 5, 2001.

Description of amendment request: The licensee proposed to amend Section 3.6.1.3, ''Primary Containment Isolation Valves,'' of the unit's Technical Specifications (TSs). Surveillance Requirement (SR) 3.6.1.3.9 currently requires verification of the actuation capability of each excess flow check valve (EFCV) at least once per 24 months. One proposed change will result in limiting the surveillance to only those EFCVs in instrumentation lines connected to the reactor coolant pressure boundary. The requirement for testing of EFCVs other than those in reactor instrumentation lines is proposed to be relocated to a licensee-controlled document. Another proposed change is to revise the SR by allowing a representative sample of reactor instrumentation line EFCVs to be tested every 24 months, such that each reactor instrumentation line EFCV will be tested every 10 years.

The associated licensee-controlled TSs Basis document would also be changed to reflect the above TSs 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. The NRC staff has reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes to SR 3.6.1.3.9 will result in reduction in the frequency and scope of EFCV testing. No hardware design change is involved. While a postulated instrument line break accident was analyzed and evaluated as part of the design basis, no credit was given to EFCVs to limit or stop radioactive water through the ruptured instrument line. The EFCVs were not considered precursor of accidents in the unit's design basis. Accordingly, the revised scope and frequency of EFCV testing will lead to no increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No hardware design change or procedural change is involved with the proposed changes to SR 3.6.1.3.9. The amendment would only relax the frequency and scope of EFCV testing. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since no design or procedural change is involved, the proposed changes to SR 3.6.1.3.9 will not affect in any way the performance characteristics and intended functions of systems and components (i.e., the instrument lines and instruments) served by the EFCVs. Therefore, the proposed changes to SR 3.6.1.3.9 do not involve a significant reduction in a margin of safety.

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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Marsha Gamberoni.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of amendment request: February 27, 2001.

Description of amendment request: The licensee proposed to amend Technical Specifications (TSs) Section 3.3.8.2, ''Reactor Protection System (RPS) Electric Power Monitoring—Logic,'' reducing the channel calibration allowable values for overvoltage from 133.8 V to 130.2 V (for Bus A), and to 129.8 V (for Bus B). The licensee also proposed to amend Section 3.3.8.3, ''Reactor Protection System (RPS) Electric Power Monitoring—Scram Solenoids,'' reducing the channel calibration allowable values for overvoltage from 130.5 V (for Bus A) and 131.7 V (for Bus B) to 127.6 V. These proposed changes are in the conservative direction, reflecting the results of revisions to calculations to correct licensee-identified analysis deficiencies. The proposed reduced allowable values would be accompanied by an increase in channel calibration frequency from once per 24-months to once per 184 days.

The associated licensee-controlled TSs Basis document would also be changed to reflect the above TSs 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. The NRC staff has reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes to Sections 3.3.8.2 and 3.3.8.3 will be made in a conservative direction. No hardware design change is involved, thus there will be no adverse effect on the functional performance of any plant structure, system, or component (SSC). All SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. Accordingly, the revised allowable values and channel calibration frequencies will lead to no increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No hardware design change or procedural change is involved with the proposed changes to these sections. The amendment does not involve any changes in design or performance of any SSC; all SSCs will continue to perform as previously analyzed by the licensee and previously accepted by the staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since no design or procedural change is involved, the proposed changes to Sections 3.3.8.2 and 3.3.8.3 will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment requests: October 30, 2000.

Description of amendment requests: The proposed amendments would allow modification of the eight double-leaf doors in the auxiliary building special ventilation zone. These doors serve as ''blowout panels'' in case of a high-energy line break (HELB) accident inside the auxiliary building. Currently, these doors are held in place by the resistance from the hinges and door center latch. The licensee proposes to install additional ''breakaway'' pins on these doors to increase the restraining forces upon these doors to minimize nuisance alarms from these doors. However, the licensee has determined that this modification did not meet the criteria of 10 CFR 50.59 and therefore requires prior NRC staff review and approval. These amendments do not involve changes to the Operating Licenses or the 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:

1. Does operation of the facility with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not significantly affect any system that is a contributor to initiating events for previously evaluated accidents. The addition of a ceramic latch pin in selected Auxiliary Building Special Ventilation Zone (ABSVZ) boundary doors will provide a small restraining force to hold the doors closed under typical operating conditions, but will snap under the pressures produced on the doors by a high-energy line break, thus allowing the doors to swing open and provide a relief path for steam discharge into the Auxiliary Building compartments during a HELB. Testing has established that the ceramic pins will breakaway under a load that is significantly lower than the differential pressure loading on the boundary doors assumed in the HELB analyses. In addition, improving the ability to keep these doors closed under normal operating conditions helps to assure maintenance of the ABSVZ boundary integrity assumed in the LOCA [loss-of-coolant accident] and offsite dose analyses. Thus it is concluded that the proposed changes do not involve any significant increase in the probability or consequence of an accident previously evaluated.

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

While the proposed modification alters the design of plant equipment, it does not alter the function or the manner of operation [of] any plant component and does not install any new or different equipment. During a HELB selected ABSVZ boundary doors are required to swing open to provide a steam relief path. The use of ceramic pins to restrain these doors against inadvertent opening during normal operations does not alter the accident mitigation function of these doors. Testing has established that these ceramic pins will break before the pressure in the Auxiliary Building reaches the relief point assumed in the HELB analyses. This situation does not create the possibility of a new or different kind of accident from those previously analyzed.

3. Does operation of the facility with the proposed amendment involve a significant reduction in a margin of safety?

Because testing has established that these ceramic pins will break before the pressure in the Auxiliary Building reaches the relief point assumed in the HELB analyses, the accident mitigation function of the ABSVZ boundary doors will be preserved. In the event of a HELB the ABSVZ boundary doors will swing open and provide a steam relief path. Thus avoiding any increased Auxiliary Building compartment pressures that might challenge the requirements on ventilation boundary leakage and block wall structural integrity established to maintain assurance of control room habitability.

Thus, the proposed change does not involve a significant reduction in the margin of safety associated with the safety limits inherent in either the principle barriers to a radiation release (fuel cladding, RCS [reactor coolant system] boundary, and reactor containment), or the maintenance of critical safety functions (subcriticality, core cooling, ultimate heat sink, RCS inventory, RCS boundary integrity, and containment integrity).

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

NRC Section Chief: Claudia M. Craig.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: February 14, 2001.

Description of amendment request: The proposed amendment would make minor changes to the Ginna Improved Technical Specifications (ITS) format to allow for maintaining, viewing, and publishing them with different software package. The proposed amendment would also revise the ITS section 5.5.13, ''Technical Specifications Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) dated October 4, 1999.

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:

Evaluation of Administrative Formatting Changes

The administrative changes associated with the minor revisions in the Ginna Station ITS format to allow for maintaining, viewing, and publishing them with different software package do not involve a significant hazards consideration as discussed below:

(1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes involve minor reformatting of the existing Improved Technical Specifications to provide compatibility with the software package that is proposed for maintenance of the electronic ITS files and do not include any technical issues. As such, these changes are administrative in nature and do not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of safety because the changes do not impact any safety analysis assumptions. These changes are administrative in nature. As such, no question of safety is involved, and the changes do not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Evaluation of Administrative 10 CFR 50.59 Changes

The administrative changes associated with the revision to ITS section 5.5.13, ''Technical Specifications (TS) Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 do not involve a significant hazards consideration as discussed below:

(1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC [Nuclear Regulatory Commission] with the revision of 10 CFR 50.59. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a

margin of safety. The proposed changes will not reduce a margin of safety because the changes do not impact any safety analysis assumptions. Changes to the ITS Bases that result in meeting the criteria in paragraph 10 CFR 50.59(c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. As such, no question of safety is involved, and the changes do not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

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: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Marsha Gamberoni.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of amendment request: February 20, 2001.

Description of amendment request: The proposed license amendment would eliminate the security plan requirements from the 10 CFR Part 50 licensed site after the Rancho Seco spent nuclear fuel has been transferred from the spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI). Specific changes would include deleting Section 2.C(3) ''Physical Protection'' from Rancho Seco Facility Operating License No. DPR–54 and deleting all references in the Permanently Defueled Technical Specifications to the Rancho Seco Nuclear Generating Station security plans.

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 evaluated?

No. The physical structures, systems, and components of the Rancho Seco 10 CFR 50 licensed site and the operating procedures for their use are unaffected by the proposed change. The elimination of the security requirements from the 10 CFR Part 50 licensed site does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility.

Elimination of the security requirements for the 10 CFR Part 50 license is predicated upon completion of the transfer of all nuclear fuel from the spent fuel pool to the ISFSI. The planned 10 CFR 72 licensing controls for the ISFSI will provide adequate confidence that personnel and equipment can perform satisfactorily for normal operations of the ISFSI and respond adequately to off-normal and accident events. The Rancho Seco Physical Protection Plan (PPP) will also provide confidence that security personnel and safeguards systems will perform satisfactorily to ensure adequate protection for the storage of spent nuclear fuel. Therefore, the proposed 10 CFR Part 50 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed change is security related and has no direct impact on plant equipment or the procedures for operating plant equipment. The safety analysis for the facility remains complete and accurate. There are no physical changes to the facility, and the plant conditions for which the design basis accidents have been evaluated are still valid.

Because the ISFSI site is segregated from the 10 CFR Part 50 licensed site, licensed security activities under the 10 CFR Part 50 license will no longer be necessary after all the nuclear fuel has been moved. The planned 10 CFR 72 licensing controls for the ISFSI will provide adequate confidence that personnel and equipment can perform satisfactorily for normal operations of the ISFSI and respond adequately to off-normal and accident events. Moreover, the ISFSI will be physically separate from the 10 CFR 50 licensed site structures and equipment. Therefore, the proposed 10 CFR Part 50 license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. As described above, the proposed change is security related and has no direct impact on plant equipment or the procedures for operating plant equipment. There are no changes to the design or operation of the facility.

The assumptions for fuel handling and other accidents are not affected by the proposed license amendment. Accordingly, neither the design basis nor the accident assumptions in the Defueled Safety Analysis Report (DSAR), nor the PDTS Bases are affected. 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 request involves no significant hazards consideration.

Attorney for licensee: Dana Appling, Esq., Sacramento Municipal Utility District, P.O. Box 15830, Sacramento, California 95852–1830.

NRC Section Chief: Stephen Dembek.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: May 17, 2000, as supplemented by letters dated August 31, 2000, and January 31, 2001.

Brief description of amendments: The proposed amendments would revise the Allowable Values specified in Technical Specification (TS) Table 3.3.5–1, ''Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation'' to ensure that the 6.9 kiloVolt (kV) and 480 Volt (V) undervoltage relays initiate the necessary actions when required. In addition, a proposed administrative change to Condition D of TS 3.3.5, would eliminate the term ''undervoltage,'' consistent with the proposed changes to TS Table 3.3.5–1.

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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

The proposed License Amendment Request includes more restrictive Allowable Values for the Preferred offsite source bus undervoltage function, the Alternate offsite source bus undervoltage function, the 6.9 kV Class 1E bus loss of voltage function, the 6.9 kV Class 1E bus degraded voltage function and the 480 V Class 1E bus degraded voltage function. These more restrictive values assure that all applicable safety analysis limits are being met. The 480 V low grid undervoltage relay allowable value is being lowered to the same as the 480 V degraded voltage relays which matches its function. This is a less restrictive value but the value still assures that all applicable safety analysis limits are being met. Lowering of the 480 V low grid undervoltage allowable value will minimize unnecessary actuations that could challenge plant systems. Changing the 6.9 kV and 480 V degraded voltage, 480 V low grid undervoltage, the 6.9 kV loss of voltage, and the preferred and alternate bus undervoltage

Allowable Values in the TSs has no impact on the probability of occurrence of any accident previously evaluated. Because all accident analyses continue to be met, these changes do not impact the consequences of any accident previously evaluated.

Removal of the lower limit for the 6.9 kV Class 1E bus loss of voltage relays does not impact the probability of occurrence of any accident previously evaluated. None of the accident analyses are affected; therefore, the consequences of all previously evaluated accidents remain unchanged.

The proposed administrative change to Condition D of TS 3.3.5, which would eliminate the term ''undervoltage,'' consistent with the proposed changes to TS Table 3.3.5–1 is administrative in nature. None of the accident analyses are affected; therefore, the probability and consequences of all previously evaluated accidents remain unchanged.

None of the changes to TS Table 3.3.5–1 affect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed administrative change to Condition D of TS 3.3.5, which would eliminate the term ''undervoltage,'' consistent with the proposed changes to TS Table 3.3.5–1 is administrative in nature. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

There were no changes made to any of the accident analyses or safety analysis limits as a result of this proposed change. Further, the proposed change does not affect the acceptance criteria for any analyzed event. Removal of the lower limit for the 6.9 kV Class 1E bus loss of voltage relays does not change the margin of safety. Each allowable value, as revised, assures the safety analysis limits assumed in the safety analyses as discussed in Chapter 15 of the Final Safety Analysis Report is maintained. The margin of safety established by the Limiting Conditions for Operation also remains unchanged. Thus there is no effect on the margin of safety.

The proposed administrative change to Condition D of TS 3.3.5, which would eliminate the term ''undervoltage,'' consistent with the proposed changes to TS Table 3.3.5–1 is administrative in nature. Thus there is no effect on the margin of safety.

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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of application request: February 15, 2001 (ULNRC–4391).

Description of amendment request: The proposed amendment would delete paragraph d.1.j(2) in Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' that requires all SG tubes containing an Electrosleeve, a Framatome proprietary process, to be removed from service within two operating cycles following installation of the first Electrosleeve. This requirement was incorporated in TS 5.5.9 in Amendment No. 132 issued May 21, 1999. The first Electrosleeve tube was installed in the Fall of 1999 and the two-cycle allowance will expire in the Fall of 2002.

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 would remove the restriction that requires all steam generator tubes repaired with Electrosleeves to be removed from service at the end of two operating cycles following installation of the first Electrosleeve. This would allow all steam generator tubes repaired with Electrosleeves to remain in service. Reference 2 [licensee's letter dated October 27, 1998] concluded that there was no significant increase in the probability or consequences of an accident previously evaluated when using the Electrosleeve repair method. The two operating cycle restriction was invoked because the NRC staff concluded that the UT [ultrasonic] methods used to perform NDE [nondestructive examination] for inservice inspections of the Electrosleeved tubes could not reliably depth size stress corrosion cracks to ensure that structural limits are maintained.

Revision 4 to topical report BAW–10219P [nonproprietary version is attached to the application] has addressed the concerns that resulted in the restriction of two operating cycles and consequently, the probability of an accident previously evaluated is not significantly increased. As a result, the consequences of any accident previously evaluated are not affected.

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. Reference 2 concluded that the use of the Electrosleeve repair method did not create the possibility of a new or different kind of accident from any accident previously evaluated when using this method to repair steam generator tubes. This proposed change removes the two operating cycle limit for the Electrosleeved tubes based on the evaluations and justifications of the NDE techniques used to perform inservice examinations of the Electrosleeved steam generator tubes provided in Revision 4 of the topical report.

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

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

The proposed change does not affect the acceptance criteria for an analyzed event. The margin of safety presently provided by the structural integrity of the steam generator tubes remains unchanged. Reference 2 concluded that the use of the Electrosleeve repair method did not involve a significant reduction in a margin of safety when using this method to repair steam generator tubes. The proposed change removes the two operating cycle limit based on the evaluations and justifications presented in Revision 4 of the topical report.

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

The reference to ''Reference 2'' in the criteria above is a reference to the licensee's letter dated October 27, 1998, and the no significant hazards consideration (NHSC) in that letter, which was published in the **Federal Register** (63 FR 66604) on December 2, 1998. This NHSC is applicable to the current application because it applies to the use of Electrosleeved steam generator tubes, the subject of the current application.

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 O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Units No. 1 and No. 2, Surry County, Virginia

Date of amendment request: December 7, 2000. This amendment request supersedes the November 29, 1999, request in its entirety. The November 29, 1999, request was noticed on March 22, 2000 (65 FR 15388).

Description of amendment request: The proposed changes will modify the Technical Specifications (TS) in Section 3.23 for the Main Control Room and Emergency Switchgear Room Ventilation and Air Conditioning Systems; TS Surveillance Requirement Section 4.20 for the Control Room Air Filtration System; and TS Surveillance Requirement Section 4.12 for the Auxiliary Ventilation Exhaust Filter Trains. The proposed changes will revise the above Surveillance Requirements for the laboratory testing of the carbon samples for methyl iodide removal efficiency to be consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989, ''Standard Test Method for Nuclear-Graded Activated Carbon,'' with qualification as the laboratory testing standard for both new and used charcoal adsorbent used in the ventilation 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:

Criterion 1—Operation of Surry Units 1 and 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes only modify surveillance testing requirements and do not affect plant systems or operation and therefore do not increase the probability or the consequences of an accident previously evaluated. The proposed surveillance requirements adopt ASTM D–3803–1989, with qualification, as the laboratory method for testing samples of the charcoal adsorber for methyl iodide removal efficiency consistent with NRC's Generic Letter 99–02. This method of testing charcoal adsorbers provides an acceptable approach for determining methyl iodide removal efficiency and ensuring that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle. There is no change in the method of plant operation or system design with this change.

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

The proposed changes only modify surveillance testing requirements and do not impact plant systems or operations and therefore do not create the possibility of an accident or malfunction of a different type than evaluated previously. The proposed surveillance requirements adopt ASTM D3803–1989, with qualification, as the laboratory method for testing samples of the charcoal adsorber for methyl iodide removal efficiency. This change is in response to NRC's request in Generic Letter 99–02. There is no change in the method of plant operation or system design. There are no new or different accident scenarios, transient precursors, nor failure mechanisms that will be introduced.

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

The proposed changes only modify surveillance test requirements and do not impact plant systems or operations and therefore do not significantly reduce the margin of safety. The revised surveillance requirements adopt ASTM D3803–1989, with qualification, as the laboratory method for testing samples as the charcoal adsorber for methyl iodide removal efficiency. The 1989 edition of this standard imposes stringent requirements for establishing the capability of new and used activated carbon to remove methyl iodide from air and gas streams. The results of this test provide a more conservative estimate of the performance of nuclear-graded activated carbon used in nuclear power plant HVAC systems for the removal of methyl iodide. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle.

This evaluation concludes that the proposed amendment to the Surry Units 1 and 2 Technical Specifications does not involve a significant increase in the probab[ility] or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief (Acting): M. Banerjee.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: December 12, 2000, as supplemented January 8 and February 22, 2001.

Description of amendment request: The proposed changes would revise Technical Specification (TS) 3.17.4 and 3.17.5 and the appropriate Bases. The proposed changes will acknowledge the establishment of seal injection for the reactor coolant pump in an isolated and drained loop as a prerequisite for the vacuum-assisted backfill technique. Also, the proposed changes include additional limiting conditions for operation and surveillance requirements for the sources of borated water used during loop backfill, and revised reactivity controls for an isolated-filled loop.

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 Technical Specification limiting conditions for operation and surveillance requirements ensure that the initiation of seal injection in order to allow a partial vacuum to be established in an isolated and drained loop will not create the potential for an inadvertent/undetected introduction of under-borated water into an isolated loop prior to returning the isolated loop to service. The proposed Technical Specification controls prevent any additions of makeup or seal injection that would violate the existing shutdown margin requirements for the active portion of the Reactor Coolant System. Thus, adequate Technical Specification controls are established to preclude an inadvertent/undetected positive reactivity addition event. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

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

There are no modifications to the plant as a result of the changes. The proposed Technical Specification Limiting Conditions for Operation and Surveillance Requirements ensure that the initiation of seal injection will not create an undetected positive reactivity addition. No new accident or event initiators are created by the initiation of seal injection for the RCP [reactor coolant pump] in the isolated loop in order to establish a partial vacuum in that isolated and drained loop. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Does the change involve a significant reduction in the margin of safety as defined in the bases on any Technical Specifications.

The proposed changes have no effect on safety analyses assumptions. Rather, the proposed changes acknowledge the establishment of seal injection for the RCP in the isolated and drained loop as a prerequisite for the vacuum-assisted backfill technique. The proposed Technical Specification Limiting Conditions for Operation and Surveillance Requirements ensure that the initiation of seal injection in order to allow a partial vacuum to be established in an isolated and drained loop will not create the potential for an inadvertent/undetected introduction of under-borated water into an isolated loop prior to returning the isolated loop to service. Adequate Technical Specifications controls are established to preclude an inadvertent/undetected positive reactivity addition event. In addition, the proposed controls prevent any additions of makeup or seal injection that would violate the existing shutdown margin requirements for the active portion of the Reactor Coolant System. Therefore, the proposed changes do not result in 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief (Acting): M. Banerjee.

Previously Published Notices of Consideration of Issuance of Amendments 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 was the same as above. They were published as individual notices either because 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 consideration.

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.

Tennessee Valley Authority, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of application for amendments: February 5, 2001 (TS–413).

Brief description of amendments: Changes the Reactor Vessel Material Surveillance schedule to allow a one-cycle delay in removal of the second capsule.

Date of publication of individual notice in the ***Federal Register:*** February 28, 2001 (66 FR 12818).

Expiration date of individual notice: March 30, 2001.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Exelon Generation Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: September 5, 2000, as supplemented January 17, 2001.

Brief description of amendments: The amendments revised Surveillance Requirement 4.6.3.4 to allow a representative sample of reactor instrumentation line excess flow check valves (EFCVs) to be tested every 24 months, instead of testing each EFCV every 24-months.

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

Effective date: February 23, 2001.

Amendment Nos.: 148 and 110.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2021).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 2001.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: November 9, 2000.

Brief description of amendment: By letter dated November 9, 2000, FirstEnergy Nuclear Operating Corporation (FENOC), requested a Technical Specification change for Davis-Besse Nuclear Power Station (DBNPS), Unit 1. The proposed Technical Specification (TS) changes would relocate Technical Specification 3/4.4.9.2, Reactor Coolant System—Pressurizer, to the Davis-Besse Nuclear Power Station (DBNPS) Technical Requirements Manual (TRM). The TRM is a DBNPS controlled document which has been incorporated into the Davis-Besse Updated Safety Analysis Report (USAR).

Date of issuance: February 27, 2001.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 245.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81919).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 5, 2000, as supplemented by letter dated January 15, 2001.

Brief description of amendment: This amendment implements technical specification (TS) changes associated with thermo-hydraulic stability monitoring. New TS 3.3.1.3, ''Oscillation Power Range Monitor

(OPRM) Instrumentation,'' is added, providing the minimum operability requirements for the OPRM channels, the Required Actions when they become inoperable, and appropriate surveillance requirements. The amendment also removes monitoring guidance from TS 3.4.1, ''Recirculation Loops Operating,'' that will no longer be necessary due to the activation of the OPRM instrumentation, and updates TS 5.6.5, ''Core Operating Limits Report (COLR),'' to require the applicable setpoints for the OPRMs to be included in the COLR.

Date of issuance: February 26, 2001.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 118.

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

Date of initial notice in ***Federal Register:*** May 31, 2000 (65 FR 34745).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 1, 2000.

Brief description of amendment: The Technical Specification (TS) Section 3.4.14, ''RCS Leak Detection Instrumentation, Surveillance Requirements,'' was changed to extend the calibration interval of the containment sump monitor to 24 months.

Date of issuance: March 7, 2001.

Effective date: March 7, 2001.

Amendment No.: 195.

Facility Operating License No. DPR–72: Amendment revised the TSs.

Date of initial notice in ***Federal Register:*** July 12, 2000 (65 FR 43048).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2001.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: October 30, 2000.

Brief description of amendment: This amendment revises Technical Specification (TS) Limiting Condition For Operation 3.9.4.b to allow both doors of the containment personnel airlock to be open during core alterations if: (1) at least one personnel airlock door is capable of being closed, (2) the plant is in Mode 6 with at least 23 feet of water above the fuel in the reactor core, and (3) a designated individual is available outside the personnel airlock to close the door.

Date of Issuance: February 27, 2001.

Effective Date: February 27, 2001.

Amendment No.: 172.

Facility Operating License No. DPR–67: Amendment revised the TS.

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81920).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 2001.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc. and Saxton Nuclear Experimental Corporation, Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania

Date of application for amendment: November 30, 2000 and supplemented on January 18, 2001.

Brief description of amendment: The amendment changes the Amended Facility License to reflect the change in the legal name of GPU Nuclear Corporation to GPU Nuclear, Inc. wherever it appears in the license.

Date of Issuance: March 8, 2001.

Effective date: The license amendment is effective as of its date of issuance.

Amendment No.: 17.

Amended Facility License No. DPR–4: The amendment revised the Amended Facility License.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2010). The Commission's related evaluation of the amendment is contained in a safety evaluation dated March 8, 2001.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: November 5, 1999, as supplemented by electronic mail dated March 22 and letter dated September 28, 2000.

Brief description of amendment request: The amendment revises technical specification requirements to submit biennial reports every 24-months instead of prior to March 1 of every other year. It also eliminates the requirements to notify the Nuclear Regulatory Commission (NRC) of exceeding environmental limits and changes to environmental permits such as the National Pollution Discharge Elimination System permit. The licensee's November 5, 1999, submittal proposed revising technical specifications dealing with eliminating notifying the NRC for exceeding limits of minor permits where there is no identifiable environmental or public health concerns and exceptional occurrences (unusual or important events, exceeding limit of relevant permits). Since additional information would be required to continue this part of the review, the licensee withdrew this portion of their original application dated November 5, 1999, and replaced it in its entirety with a supplemental letter dated September 28, 2000.

Date of issuance: March 1, 2001.

Effective date: Immediately, to be implemented within 120 days.

Amendment No.: 55.

Facility Operating License No. DPR–73: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 12, 2000 (65 FR 1924). The September 28, 2000, supplemental letter replaced in its entirety the licensee's original application dated November 5, 1999. The supplement did not expand the scope of the original request, nor did it change the proposed no significant hazards consideration finding. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2001.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: June 30, 2000, as supplemented on September 22 and November 20, 2000; and January 26 and February 1, 2001.

Brief description of amendment: This amendment changes the Millstone Nuclear Power Station, Unit No. 3 licensing basis. The amendment authorizes changes to the Final Safety Analysis Report (FSAR) regarding the installation of a new sump pump system in the engineered safety features building.

Date of issuance: February 26, 2001.

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

Amendment No.: 195.

Facility Operating License No. NPF–49: Amendment authorizes changes to the FSAR.

*Date of initial notice in **Federal Register:*** October 18, 2000 (65 FR 62388).

The September 22 and November 20, 2000, and January 26 and February 1, 2001, letters provided clarifying information that did not change the scope of the amendment or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: November 29, 1999, as supplemented November 10 and December 15, 2000.

Brief description of amendment: The amendment revises the Kewaunee Nuclear Plant Technical Specifications to incorporate requested changes per Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999.

Date of issuance: February 28, 2001.

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

Amendment No.: 152.

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

*Date of initial notice in **Federal Register:*** December 13, 2000 (65 FR 77921).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 2001.

No significant hazards consideration comments received: No.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: July 28, 2000, as supplemented by letter dated December 14, 2000.

Brief description of amendment: The amendment revises Sections 2.1.4, 3.1, 3.17, Table 3–13, Table 3–14, and associated Bases of the Fort Calhoun Station Technical Specifications to allow the installation of ABB Combustion Engineering leak tight sleeves as an alternative tube repair method to plugging defective steam generator tubes.

Date of issuance: March 1, 2001.

Effective date: March 1, 2001, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 195.

Facility Operating License No. DPR–40. Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 18, 2000 (65 FR 62388).

The December 14, 2000, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 30, 2000.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.5.14, ''Technical Specifications (TS) Bases Control Program'' to reflect the changes made to 10 CFR 50.59 as published in the **Federal Register** on October 4, 1999 (Volume 64, Number 191, ''Changes, Tests, and Experiments,'' pages 53582 through 53617). A conforming change is made to TS 5.5.14 to replace the word ''involve'' with the word ''require,'' as it applies to changes to the TS Bases without prior NRC approval.

Date of issuance: March 2, 2001.

Effective date: March 2, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment Nos.: Unit 1–145; Unit 2–144

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 27, 2000 (65 FR 81928)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 2, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 6, 2000.

Brief description of amendments: The amendments revised Section 5.0 of the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 Technical Specifications to change management titles from (a) ''Vice President, Diablo Canyon Operations and Plant Manager'' to ''plant manager,'' (b) ''Senior Vice President and General Manager—Nuclear Power Generation'' to ''specified corporate officer,'' (c) ''Radiation Protection Director'' to ''radiation protection manager,'' and (d) ''Operations Director'' to ''operations manager.''

Date of issuance: March 7, 2001.

Effective date: March 7, 2001, and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1–146; Unit 2–145.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 24, 2001 (66 FR 7685).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 7, 2001.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: February 29, 2000 (submitted by PP&L, Inc., the licensee before July 1, 2000).

Brief description of amendments: The amendments incorporated a reference to Supplement 3 ''Application Enhancements'' for the approved Topical Report PL–NF–90–001–A, ''Application of Reactor Analysis Methods for BWR [Boiling Water Reactor] Design and Analysis,'' into TS 5.6.5, Core Operating Limits Report.

Date of issuance: February 28, 2001.

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

Amendment Nos.: 189 and 163.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 18, 2000 (65 FR 62390).

The Commission's related evaluation of the amendments is contained in a

Safety Evaluation dated February 28, 2001.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: March 20, 2000 (submitted by PP&L, Inc., the licensee before July 1, 2000), as supplemented December 1, 2000, and January 22, 2001 (submitted by PPL Susquehanna, LLC, the licensee on and after July 1, 2000).

Brief description of amendment: The amendment revised the minimum critical power ratio safety limits.

Date of issuance: March 6, 2001.

Effective date: As of date of issuance and shall be implemented upon startup following the Unit 2 tenth refueling and inspection outage.

Amendment Nos.: 164.

Facility Operating License No. NPF–22. The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 13, 2000 (65 FR 77924).

The supplemental letters provided additional information but did not change the initial no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 2001.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: November 3, 2000, as supplemented February 1, 2001.

Brief description of amendments: The amendments revise Technical Specification 5.5.11, ''Technical Specification Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 which were published in the **Federal Register** (64 FR 53582) on October 4, 1999.

Date of issuance: March 6, 2001.

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

Amendment Nos.: 224 and 165.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 13, 2000 (65 FR 77925).

The supplement dated February 1, 2000, provided clarifying information that did not change the scope of the November 3, 2000, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 2001.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: November 16, 2000, as supplemented on January 11, 2001.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 5.5.14, ''Technical Specification Bases Control Program'' to provide consistency with the changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) dated October 4, 1999. Specifically, the amendments remove the term ''unreviewed safety question'' from TS 5.5.14.b.2. In addition, two editorial corrections are also made on page 5.5–18.

Date of issuance: March 1, 2001.

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

Amendment Nos.: 118 and 96.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 13, 2000 (65 FR 77927).

The supplemental letter dated January 11, 2001, provided clarifying information that did not change the scope of the November 16, 2000, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 2001.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket No. 50–499, South Texas Project, Unit 2, Matagorda County, Texas

Date of amendment request: February 21, 2000, as supplemented by letters dated January 24 and 30, and February 28, 2001. The January 24 and 30, and February 28, 2001 letters, provided additional clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration.

Brief description of amendments: The Amendment revises the Technical Specifications (TSs) approving the application of the 3-volt repair criteria to the methodology for repair of steam generator (SG) tubes. The new criteria will apply for Unit 2 Cycle 9 only.

Date of issuance: March 8, 2001.

Effective date: The Amendment is effective on the date of issuance.

Amendment No.: 114.

Facility Operating License No. NPF–80: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 22, 2000 (65 FR 15386).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2001.

No significant hazards consideration comments received: No.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: September 6, 2000, as supplemented by letters dated December 14, 2000, and January 25, 2001.

Brief description of amendments: The amendment changes Comanche Peak Electric Station (CPSES), Units 1 and 2, Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' to permit installation of laser welded tubes sleeves in CPSES Unit 1 steam generator as an alternative to plugging defective tubes, and TS 5.6.10, ''Steam Generator Tube Inspection Report,'' is revised to address reporting requirements for repaired tubes. Also an editorial correction is made to Table 5.5–2.

Date of issuance: February 20, 2001.

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

Amendment Nos.: 83 and 83.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 1, 2000 (65 FR 65350).

The supplemental letters dated December 14, 2000, and January 25, 2001, provided additional information that clarified the application, did not expand the scope of the application, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 20, 2001.

No significant hazards consideration comments received: No.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: December 6, 2000.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.5.14, ''Technical Specifications (TS) Bases Control Program'' and TS 5.5.17, ''Technical Requirements Manual (TRM)'' to reflect the changes made to 10 CFR 50.59 as published in the **Federal Register** on October 4, 1999 (Volume 64, Number 191, ''Changes, Tests, and Experiments,'' pages 53582 through 53617). A conforming change is made to TS 5.5.14 and 5.5.17 to replace the word ''involve'' with the word ''require,'' as it applies to changes to the TS Bases or TRM without prior Nuclear Regulatory Commission approval.

Date of issuance: March 5, 2001.

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

Amendment Nos.: 84 and 84.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 10, 2001 (66 FR 2024).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 2001.

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: March 29, 2000, as supplemented December 6, 2000, and March 1, 2001.

Brief Description of amendments: These amendments revise TS Sections 3.19 and 4.1. The changes specify the requirements for two redundant trains of bottled air, specify remedial actions when one train or both trains are inoperable, eliminate the extension of the allowed outage and remedial action time of 8 hours to 24 hours currently permitted by TS 3.19.B, specify remedial actions for an inoperable control room pressure boundary, and include additional surveillance testing requirements. The Bases sections for TS 3.19 and TS 4.1 are revised for consistency with the respective TS.

Date of issuance: March 9, 2001.

Effective date: March 9, 2001.

Amendment Nos.: 223 and 223.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48761). The December 6, 2000, and March 1, 2001, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2001.

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: December 7, 2000.

Brief description of amendment: The amendment deletes Technical Specifications (TS) Section 5.5.3, ''Post Accident Sampling System,'' for Wolf Creek Generating Station and thereby eliminates the requirements to have and maintain the post-accident sampling system. The amendment also revises TS Section 5.5.2, ''Primary Coolant Sources Outside Containment,'' to reflect the elimination of PASS.

Date of issuance: March 2, 2001.

Effective date: March 2, 2001, and shall be implemented on or before December 1, 2001.

Amendment No.: 137.

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

*Date of initial notice in **Federal Register:*** January 10, 2001 (66 FR 2026).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2001.

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: December 8, 2000.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.14, ''Technical Specifications (TS) Bases Control Program'' to reflect the changes made to 10 CFR 50.59 as published in the **Federal Register** on October 4, 1999 (Volume 64, Number 191, ''Changes, Tests, and Experiments,'' pages 53582 through 53617). A conforming change is made to TS 5.5.14 to replace the word ''involves'' with the word ''requires,'' as it applies to changes to the TS Bases without prior NRC approval.

Date of issuance: March 2, 2001.

Effective date: March 2, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 138.

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

*Date of initial notice in **Federal Register:*** January 10, 2001 (66 FR 2027).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2001.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Sinificant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By April 20, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 20555–001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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 20555–001, 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: February 1, 2001.

Description of amendment request: The amendment removes the inservice inspection requirements of Section XI of the "American Society of Mechanical Engineers Boiler and Pressure Vessel Code" from the Monticello Technical Specifications and relocates them to a licensee-controlled program.

Date of issuance: March 1, 2001.

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

Amendment No.: 116.

Facility Operating License No. (DPR–22): Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (66 FR 10535, dated February 15, 2001). 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 March 19, 2001, 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, state consultation, and final NSHC determination are contained in a Safety Evaluation dated March 1, 2001.

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

NRC Section Chief: Claudia M. Craig.

Dated at Rockville, Maryland this 13th day of March 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–6732 Filed 3–20–01; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration; (Hovnanian Enterprises, Inc., Class A Common Stock, $.01 Par Value) File No. 1–08551

March 15, 2001.

Hovnanian Enterprises, Inc., a Delaware corporation ("Issuer"), has filed an application with the Securities and Exchange Commission ("Commission"), pursuant to section 12(d) of the Securities Exchange Act of 1934 ("Act") [1] and Rule 12d2–2(d) thereunder,[2] to withdraw its Class A Common Stock, $.01 par value ("Security"), from listing and registration on the American Stock Exchange ("Amex").

The Issuer has applied to have its Security listed on the New York Stock Exchange ("NYSE"). The NYSE approved such application on March 8, 2001. Trading in the Security is expected to commence on the NYSE, and to cease on the Amex, at the opening of business on March 15, 2001.

The Issuer has stated in its application that it has complied with the rules of the Amex governing the withdrawal of its Security and that the application relates solely to the withdrawal of the Security from listing on the Amex and shall have no effect upon its listing on the NYSE or its registration under section 12(b) of the Act.[3]

Any interested person may, on or before April 5, 2001, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549–0609, facts bearing upon whether the application has been made in accordance with the rules of the Amex and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

[1] 15 U.S.C. 78*l*(d).

[2] 17 CFR 240.12d2–2(d).

[3] 15 U.S.C. 78*l*(b).

For the Commission, by the Division of Market Regulation, pursuant to delegated authority.[4]

Jonathan G. Katz,

Secretary.

[FR Doc. 01–6951 Filed 3–20–01; 8:45 am]

BILLING CODE 8010–01–M

[4] 17 CFR 200.30–3(a)(1).

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–44071; File No. SR–PCX–01–08]

Self-Regulatory Organizations; Notice of Filing and Order Granting Accelerated Approval of a Proposed Rule Change by the Pacific Exchange, Inc. Relating to a Rebate of Marketing Charges to Market Makers

March 13, 2001.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 ("Act")[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on January 31, 2001, the Pacific Exchange, Inc. ("PCX" or "Exchange") filed with the Securities and Exchange Commission ("Commission" or "SEC") the proposed rule change as described in Items I and II below, which Items have been prepared by the PCX. The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons and to grant accelerated approval of the proposal.

I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The PCX proposes to rebate to Market Makers on a quarterly basis the marketing charges that have not been paid to order flow providers. The text of the proposed rule change is available at the principal offices of the PCX and at the Commission.

II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the PCX included statements concerning the purpose of and basis for the proposed rule change and discussed any comments it received on the proposed rule change. The text of these statements may be examined at the places specified in Item III below. The PCX has prepared summaries, set forth in sections A, B,

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.