above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: March 29, 2000.

Howard J. Larson,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–8340 Filed 4–4–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of April 3, 10, 17, 24, May 1 and 8, 2000.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

Matters To Be Considered

Week of April 3

Thursday, April 6
8:30 a.m. Briefing by the Executive Branch (Closed—Ex. 1).

Week of April 10—Tentative

There are no meetings scheduled for the Week of April 10.

Week of April 17—Tentative

There are no meetings scheduled for the Week of April 17.

Week of April 24—Tentative

There are no meetings scheduled for the Week of April 24.

Week of May 1—Tentative

Tuesday, May 2
9:30 a.m. Briefing on Oconee License Renewal (Public Meeting) (Contact: Dave Lange, 301–415–1730).
Wednesday, May 3
9:25 a.m. Affirmation Session (Public Meeting) (if needed).
9:30 a.m. Briefing on Efforts Regarding Release of Solid Material (Public Meeting) (Contact: Frank Cardile, 301–415–6185).

Week of May 8—Tentative

Monday, May 8
10:00 a.m. Briefing on Lessons Learned from the Nuclear Criticality Accident at Tokaimura and the Implications on the NRC's Program (Public Meeting) (Contact: Bill Troskoski, 301–415–8076).
Tuesday, May 9
8:55 a.m. Affirmation Session (Public Meeting) (if needed).
9:00 a.m. Meeting with Stakeholders on Efforts Regarding Release of Solid Material (Public Meeting) (Contact: Frank Cardile, 301–415–6185).

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (Recording)—(301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

Additional Information

By a vote of 5–0 on March 30, the Commission determined pursuant to U.S.C. 552b(e) and ¶9.107(a) of the Commission's rules that ''Affirmation of (a) Petition for Leave to Intervene in Proceeding Regarding Commonwealth Edison Request for Exemption at Zion Facility; and, (b) International Uranium (USA) Corporation Commission Affirmation of Presiding Officer Decisions Denying Envirocare's Petitions for Intervention'' be held on March 30, and on less than one week's notice to the public.

The NRC Commission Meeting Schedule can be found on the Internet at:

http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov

Dated: March 31, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–8429 Filed 4–3–00; 10:53 am]

BILLING CODE 7590–01–M

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 March 11 through March 24, 2000. The last biweekly notice was published on March 22, 2000 (65 FR 15375).

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

By May 5, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment's request: February 15, 2000.

Description of amendment's request: The proposed amendments would revise the technical specifications to permit use of the Westinghouse core monitoring and support system known as Best Estimate Analyzer for Core Operations Nuclear (BEACON).

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?

Power Distribution Monitoring System (PDMS) performs continuous core power distribution monitoring. It in no way provides any protection or control system functionality. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

Continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (*i.e.,* earlier determination of an adverse condition developing) for operator action prior to having any adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

Each accident analysis addressed in the Byron and Braidwood Stations' UFSAR will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC approved reloaddesign methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be performed in accordance with the requirements set forth in 10 CFR 50.59, ''Changes, tests and experiments,'' will ensure that future reloads will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change, therefore, does 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 of accident from any accident previously evaluated?

As stated previously, the implementation of the PDMS system has no influence or impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with PDMS does not result in a change to the design basis of any plant component or system. The evaluation of the effects of the PDMS changes shows that all design standards and applicable safety criteria limits are met. These changes, therefore, do not cause the initiation of any accident nor create any new failure mechanisms. All equipment important to safety will operate as designed. Component integrity is not challenged. The proposed changes do not result in any event previously deemed incredible being made credible. The PDMS changes will not result in more adverse conditions and will not result in any increase in the challenges to safety systems. The cycle specific variables required by the PDMS are calculated using NRC approved methods. The Technical Specifications (TS) will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded.

The proposed change, therefore, does 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 the margin of safety?

The margin of safety is not affected by the implementation of PDMS. The margin of safety presently provided by current TS remains unchanged. Appropriate measures exist to control the values of these cycle-specific limits. The proposed changes continue to require operation within the core limits that are based on NRC approved reload design methodologies. The proposed changes continue to ensure that appropriate actions will be taken if limits are violated. These actions remain unchanged. The development of the reload specific limits, including Relaxed Axial Offset Control (RAOC) bands, for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload involves a 10 CFR 50.59, ''Changes, tests and experiments,'' safety review to assure that operation of the units, within the cycle-specific limits, will not involve a reduction in margin of safety.

The proposed changes, therefore, do not impact the operation of the Byron and Braidwood Stations in any manner that involves a reduction in the margin of safety.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: February 18, 2000.

Description of amendment request: The proposed amendments would remove the anticipatory reactor scram signal for turbine electro-hydraulic control (EHC) low oil pressure trip from the reactor protection system (RPS) trip function.

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 of occurrence or consequences of an accident previously evaluated?

The proposed changes remove the ''Turbine Electro-Hydraulic Control (EHC) Control Oil Pressure-Low'' scram function and the associated Limiting Safety System Setting (LSSS). The purpose of the Turbine EHC Control Oil Pressure scram is to anticipate the pressure transient which would be caused by imminent control valve fast closure on loss of control oil pressure. This function does not serve as an initiator for any accidents evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). In addition, this trip function is not credited in any design basis event and is functionally redundant to the Turbine Control Valve Fast Closure RPS trip function during a loss of EHC control oil. The Turbine Control Valve Fast Closure will initiate a scram on a loss of control oil event coincident with turbine control valve closure.

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

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

The removal of this function does not represent a change in operating parameters or introduce a new mode of operation. The pressure switches associated with the Turbine Control Valve Fast Closure function provide equivalent protection from a loss of EHC oil. For this reason, the 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?

Operation with the proposed changes in place will not change any plant operating parameters, nor any protective system actuation setpoints other than removal of the Turbine EHC Control Oil Pressure-Low scram function. The scram function associated with the Turbine Control Valve Fast Closure provides equivalent protection for events involving turbine control valve fast closure including the loss of EHC control oil pressure. For this reason, eliminating the EHC Control Oil Pressure-Low scram function, which is redundant to other protective instrumentation, does not reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: February 23, 2000.

Description of amendment request: The proposed amendments would change the pressure-temperature (P–T) limits by revising the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel (RPV) to a maximum of 32 Effective Full Power Years (EFPY).

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 of occurrence or consequences of an accident previously evaluated?

The proposed changes do not modify the reactor coolant pressure boundary, do not make changes in operating pressure, materials or seismic loading. The proposed changes adjust the reference temperature for the limiting beltline material to account for radiation effects and provide the same level of protection as previously evaluated. The proposed changes do not adversely affect the integrity of the Reactor Coolant System (RCS) such that its function in the control of radiological consequences is affected. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Dresden Nuclear Power Station. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Use of the revised P–T curves will continue to provide the same level of protection as was previously reviewed and approved.

Further, the proposed changes to the P–T curves do not affect any activities or equipment, and are not assumed in any safety analysis to initiate any accident sequence for Dresden Nuclear Power Station. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes reflect an update of the P–T curves to extend the RPV operating limit to 32 Effective Full Power Years (EFPYs). The revised curves are based on the latest American Society of Mechanical Engineers (ASME) guidance and actual operational data for the units. These proposed changes are acceptable because the ASME guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME Section XI Appendix G was approved in 1974. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: February 29, 2000.

Description of amendment request: The proposed amendments would revise the pressure-temperature (P–T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the reactor pressure vessel (RPV). The proposed changes replace the current RPV P–T limit curves with three recalculated curves that are applicable to 32 effective full power years.

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 of occurrence or consequences of an accident previously evaluated?

The proposed changes to the LaSalle County Station reactor pressure vessel (RPV) pressure-temperature (P–T) limits do not modify the boundary, operating pressure, materials or seismic loading of the rector coolant system. The proposed changes do adjust the P–T limits for radiation effects to ensure that the RPV fracture toughness is consistent with analysis assumptions and NRC regulations. 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 system such that its function in the control of radiological consequences is affected. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

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

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

Appendices G, ''Fracture Toughness Requirements,'' and H, ''Reactor Vessel Material Surveillance Program Requirements,'' of 10 CFR 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in establishing P–T limits. Appendix G of 10 CFR 50 specifies fracture toughness and testing requirements for reactor vessel material in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code and that the beltline material in the surveillance capsules be tested in accordance with Appendix H of 10 CFR 50. Appendix G also requires the prediction of the effects of neutron irradiation on the vessel embrittlement. Generic Letter 88–11, ''NRC Position on Radiation Embrittlement of Reactor Vessel Materials And Its Impact on Plant Operations,'' requests that the methods in Regulatory Guide 1.99, Revision 2, ''Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Material,'' be used to predict the effect of neutron irradiation on the reactor vessel material.

The current P–T limits for LaSalle County Station were approved by the NRC in Amendment No. 71 for Unit 1 and Amendment No. 55 for Unit 2. The NRC approval of the current pressure-temperature limits was based on their conformance to the requirements of Appendices G and H of 10 CFR 50. The NRC also noted that current P–T limits satisfied Generic Letter 88–11 because the method in Regulatory Guide 1.99, Revision 2 was used to calculate the Adjusted Reference Temperature (ART).

The methodology used to generate the revised P–T limits in the proposed changes is similar to the methodology used to generate the currently approved P–T limits, in conformance with the requirements of Appendices G and H of 10 CFR 50, consistent with the methods of Regulatory Guide 1.99, Revision 2, and consistent with the calculations contained in our July 14, 1999, proposed TS change for power uprate operation. These proposed changes are acceptable because the ASME B&PV Code guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME B&PV Code Section XI, ''Rules for Inservice Inspection of Nuclear Power Plant Components,'' Appendix G was approved in 1974. Thus, 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, PO Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

Consolidated Edison Company of New York, Inc., Docket No. 50–003, Indian Point Nuclear Generating Station, Unit 1, Buchanan, New York

Date of application for amendment: February 14, 2000.

Description of amendment request: The proposed amendment would revise Technical Specifications Sections 2.10.2, 3.1.2, 3.2.1, 4.1.8.1.b, and 4.1.8.1. Specifically, Sections 3.1.2, 3.2.1, and 4.1.8.1.b, are organizational title changes that are administrative in nature and reflect a streamlining of the Consolidated Edison Company of New York, Inc.'s, management structure. Section 4.1.8.1 is changed to reference the current sections of Part 20 of Title 10 of the *Code of Federal Regulations* (10 CFR) and to remove any ambiguity that may exist by referring to obsolete sections of the regulations. A footnote was moved from Section 2.11 to Section 2.10.2.6 to improve the clarity of the Technical Specification since it pertains to text in subsection 2.10.2.4.

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

(a) Changes to Sections 3.1.2, 3.2.1, and 4.1.8.1.b To Reflect Organizational Title Changes

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

No. The proposed change is administrative in nature. The changes involve updating Sections 3.2.1.h and 4.1.8.b to use the title ''Shift Manager'' instead of ''Senior Watch Supervisor'' and updating Section 3.1.2 and 3.1.2.b to use the title ''Plant Manager'' instead of ''General Manager—Nuclear Power Generation'' and movement of the footnote, ''*Licensed Operator for IP2.'' These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

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

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

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

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

(b) Change to Section 4.1.8.1 to Reference the Current Sections of 10 CFR 20

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

No. The proposed change [to Section 4.1.8.1] is administrative in nature. The change involves updating Section 4.1.8.1 to reference 10 CFR 20.1601(a) and 10 CFR 20.1601(b). This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

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

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

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

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

The Nuclear Regulatory Commission (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: Brent L. Brandenburg, Esq., Consolidated Edison Co. of New York, Inc., 4 Irving Place–1830, New York, NY 10003.

NRC Section Chief: Michael Masnik.

Consolidated Edison Company of New York, Inc., Docket Nos. 50–003, 50–247 Indian Point Nuclear Generating Station, Units 1 and 2, Buchanan, New York

Date of application for amendment: February 14, 2000.

Description of amendment request: The proposed amendment to the Indian Point Nuclear Generating Station, Unit Nos. 1 and 2, Environmental Technical Specifications (ETS) would change Section 5.4.1, eliminating the discussions of Section 4.2. Specifically, in ETS Section 5.4.1, Routine Reports, the proposed change seeks to delete the reference to and discussions about Section 4.2, which was deleted from the Unit 2 Operating License as part of Amendment #90. The change is administrative in nature and improves the clarity of the ETS by eliminating the reference to a section that no longer exists.

Basis for proposed no significant hazards 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 in the consequences of an accident previously evaluated?

No. The proposed change is administrative in nature. The change involves deleting, in Section 5.4.1, the reference to and the discussions about Section 4.2, which no longer exists. The monitoring requirements specified in the current Environmental Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

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

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

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

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

The Nuclear Regulatory Commission (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: Brent L. Brandenburg, Esq., Consolidated Edison Co. of New York, Inc., 4 Irving Place-1830, New York, NY 10003.

NRC Section Chief: Michael Masnik.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: November 22, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Sections 3.7.B.1 and 3.7.B.2 to reference American Society for Testing and Materials (ASTM) D3803–1989 for testing charcoal samples from the standby gas treatment system (SGTS) and the control room high efficiency air filtration systems (CRHEAFS).

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 review is presented below:

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

The accident analyses performed to ensure compliance with the dose limits of 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC 19, use assumptions regarding SGTS and CRHEAFS performance. The analyses assume SGTS train efficiency for radioiodine removal of 99% and CRHEAFS train efficiency of 95%. They also assume individual charcoal bank efficiencies of 95%.

Obtaining charcoal samples from both systems in accordance with Regulatory Position C.6.b of Regulatory Guide (RG) 1.52, Revision 2, March 1978, ensures the laboratory tests a representative sample of the activated charcoal in each system. Testing these samples in accordance with ASTM D3803–1989 at a temperature of 86 °F and 70% RH [relative humidity] ensures accurate and reproducible test results are obtained. Specifying the allowable removal efficiency as ≥97.5% ensures an appropriate safety factor is applied. This safety factor is consistent with GL 99–02. Inlet methyl iodide concentrations are specified by ASTM D3803–1989. Finally, increasing the acceptance criteria for halogenated hydrocarbon tests to 99.9% ensures system performance is consistent with accident analysis assumptions.

No accident initiators are affected by the proposed change. Increasing charcoal adsorber efficiency and reducing allowable bypass leakage ensures SGTS and CRHEAFS performance are consistent with that assumed in Pilgrim's accident analyses. Therefore, the postulated consequences are unchanged from the previously evaluated analyses.

There are no safety consequences and environmental impacts associated with the TS 5.0 pagination revision. The proposed pagination revision incorporates, in orderly fashion, pages approved by Amendments 177 and 179.

(2) The operation of Pilgrim Station 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 new or different types of accidents or malfunctions than those previously analyzed in the Updated Final Safety Analysis Report are introduced by this proposed change because there are no new failure modes being introduced. Rather, the changes being proposed reduce the possibility that existing failure modes could occur. As discussed above in the first part of this No Significant Hazards Consideration, specifying sampling and testing of charcoal adsorber banks to NRC approved standards, increasing charcoal efficiency requirements and reducing allowable bypass leakage does not challenge plant safety and will not create the possibility of a new or different kind of accident from any accident previously analyzed.

There are no safety consequences and environmental impacts associated with the TS 5.0 pagination revision. The proposed pagination revision incorporates, in orderly fashion, pages approved by Amendments 177 and 179.

(3) The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

Collecting charcoal for testing in accordance with RG 1.52 ensures a representative sample is obtained. Testing the sample in accordance with ASTM D3803–1989 at 86 °F and 70% RH ensures accurate and reproducible results are obtained. Increasing the minimum allowable charcoal efficiency from 95% to 97.5% increases the margin of safety. Increasing the minimum allowable halogenated hydrocarbon removal requirement from 99% to 99.9% also increases the margin of safety.

There are no safety consequences and environmental impacts associated with the TS 5.0 pagination revision. The proposed pagination revision incorporates, in orderly fashion, pages approved by Amendments 177 and 179.

Based on the staff's analysis, 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: W. S. Stowe, Esquire, Entergy Nuclear Generation Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Section Chief: James W. Clifford.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: March 8, 2000.

Description of amendment request: The proposed amendment would change the technical specification definition of core alteration from ''* * * the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel* * *'' to ''* * * the movement or manipulation of any fuel, sources, or reactivity control components [excluding coupling/uncoupling of CEAs [control element assemblies]] within the reactor vessel with the vessel head removed and fuel in the vessel.* * *''

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 Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The intent of the definition is to ensure that activities which could result in reactivity changes or have the potential to cause fuel damage are considered a core alteration. The current definition could be [interpreted] to apply to other activities that would not result in reactivity changes or have the potential to cause fuel damage. Thus, the modification of the definition clarifies the wording such that movement of only those components that result in reactivity changes or have the potential to cause fuel damage are specified. The modified NUREG–1432 [Standard Technical Specifications, Combustion Engineering Plants] definition was derived to limit those actions that could cause reactivity changes and potentially affect the probability or consequences of fuel handling accidents. Therefore, changing the definition of a core alteration to movement of those components that directly affect reactivity will not result in an increase in the probability or consequences associated with a fuel handling accident.

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

Criterion 2 Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed definition identifies specific components that if moved or manipulated would result in reactivity changes. The movement or manipulation of items such as lights, video cameras, and reactor vessel material specimen capsules within the reactor vessel will not result in changes in reactivity. Additionally, no reactivity change would result with the withdrawal and insertion of incore detectors or the movement of the reactor vessel upper internals within the reactor vessel with fuel in the vessel.

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

Criterion 3 Does Not Involve a Significant Reduction in the Margin of Safety

The core alteration definition is based on the need for control of reactivity changes and the consequences of fuel handling accidents. The proposed change provides clarity as to what component movement or manipulation results in reactivity changes. The proposed change is in accordance with the guidance provided in NUREG–1432 for a core alteration.

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., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas.

Date of amendment request: March 9, 2000.

Description of amendment request: The proposed amendment would revise the license as follows:

For Cycle 14 only, Entergy Operations[, Inc.] shall be permitted to operate the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria. In accordance with Principle 5 in Regulatory Guide 1.174 concerning monitoring operational experience to ensure that performance is consistent with risk predictions, if Entergy Operations plugs or repairs steam generator tubes during Cycle 14, then the steam generators shall be reinspected to the extent necessary to verify that they have been returned to a condition consistent with the risk assessment.

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 Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

A steam generator tube rupture is an accident previously evaluated in the ANO–2 [Arkansas Nuclear One, Unit 2] Safety Analysis Report. The probability of tube burst under design basis accident conditions is only slightly increased by the proposed change due to the minor reduction in margin of safety associated with tubing structural integrity, but is within the current industry guidance of NEI [Nuclear Energy Institute] 97–06, ''Steam Generator Program Guidelines.'' Detailed studies have been performed to evaluate the probable condition of the steam generator tubing for the remainder of cycle 14 operation. These studies show less than a 0.1 percent increase in the probability of tube rupture under worst case design basis accident conditions as a result of the proposed change.

This change does not modify any parameter that will increase radioactivity in the primary system or increase the amount of radioactive steam released from the secondary safety valves or atmospheric dump valves in the event of a tube rupture.

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

Criterion 2 Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The scope of this change does not establish a potential new accident precursor. The design basis accident analyses for ANO–2 include the consequences of a double-ended break of one steam generator tube which bounds other postulated failure mechanisms. The proposed change does not modify any mode of operation or modify existing periodic inservice inspection requirements.

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

Criterion 3 Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change justifies a minor reduction in the steam generator tubing structural integrity margin of safety of three times normal differential operating pressure (4050 psi). However, the margin of safety for a tube burst still remains well in excess of the 2500 psi maximum differential pressure used in the design basis accident analysis for a main steam line break. The proposed change is technically consistent with the criteria of NEI 97–06 and Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.''

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., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of amendment request: November 29, 1999, supplemented December 20, 1999.

Description of amendment request: The proposed amendment would add license condition 2.C(12) to allow a one-time extension of the steam generator inspection interval of Technical Specification 4.4.5.3.a. This would allow the steam generator inspection interval to coincide with the 8th refueling outage scheduled to begin in September 2000.

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

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

The proposed change is temporary and allows a one time extension of the steam generator (SG) surveillance requirement (SR) for Cycle 8 to allow surveillance testing to coincide with the 8th refueling outage (2R8). The proposed surveillance interval extension will not cause a significant reduction in system reliability nor affect the ability of a system to perform its design function. Current monitoring of plant conditions and the surveillance monitoring required during normal plant operation will be performed as usual to assure conformance with technical specification (TS) operability requirements.

The TS SG tube inspection is intended to prevent the ''Steam Generator Tube Failure'' analyzed in [Updated Final Safety Analysis Report] UFSAR Section 15.6.3 by maintenance of the integrity of the primary to secondary coolant boundary represented by SG tubes. The process by which this integrity is maintained is inspection of SG tubes at prescribed intervals, and the repair or removal of defective tubes from service. Inspection intervals are based on preventing corrosion growth from exceeding tube structural limits, thereby preventing tube failure. The 1998 SG inspection characterized existing tube degradation, and degraded tubes were removed from service at that time. Degradation growth rates were evaluated for the next operating interval and it was determined that the steam generator tube structural integrity is maintained. Degradation of SG tubes was prevented during the extended outage by a corrosion prevention program.

The surveillance extension does not involve a change to plant equipment and does not affect the performance of plant equipment used to mitigate an accident. This change, therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Extending the surveillance interval for the performance of specific inspections will not create the possibility of any new or different kind of accidents. No change is required to any system configurations, plant equipment or analyses.

SG tube inspections determine tube integrity and provide reasonable assurance that a tube rupture or primary to secondary leak will not occur. The only type of accident that can be postulated from extending the SG inspection interval would be a tube leak or rupture and these are analyzed in the UFSAR. No new failure modes are created by the surveillance extension. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Surveillance interval extensions will not impact any plant safety analyses since the assumptions used will remain unchanged. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since only the surveillance interval is being extended. Extending the surveillance interval for the performance of these specific inspections does not involve a significant reduction in the margin of safety derived from the required surveillances.

The margin of safety depends upon maintenance of specific operating parameters within design limits. In the case of SGs, that margin is maintained through assurance of tube integrity as the primary to secondary boundary. Assurance of tube integrity is provided through periodic in-service inspection of tubes and repair or removal of defective tubes from service. Radiation monitors provide a detection capability of primary to secondary leakage to enable a prompt response. The water chemistry of the steam generators during shutdown was maintained as described previously in Section C [Section C of Attachment B to the licensee's November 29, 1999, amendment request]. Maintenance of the SG water chemistry during power operation in accordance with Electric Power Research Institute (EPRI) guidelines provides additional margin of safety. Therefore, the plant will be maintained within the analyzed limits and the proposed extension will not significantly reduce the margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: January 19, 2000.

Description of amendment request: These proposed license amendments will revise the Technical Specifications to be consistent with the Standard Technical Specifications requirements that allow for an expanded as-found testing acceptance tolerance for the main steam safety valves (MSSV) and pressurizer code safety valves (PSV). Mode 5 operability requirements for the PSVs will also be deleted.

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

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

The probability of occurrence of an accident previously evaluated has not been increased. The changes provided in this safety evaluation do not affect the assumptions or results of any accident evaluated in the UFSAR [updated final safety analysis report]. The actual setpoints and as-left setpoint tolerances of the MSSVs and PSVs are not changed as a result of this evaluation.

Likewise, the consequences of any accident previously evaluated have not been increased. The ability of the MSSVs and PSVs to respond to accident conditions as assumed in any accident analysis has not been affected (i.e., adequate overpressure protection is provided). The proposed changes allow for the acceptance of safety valve lift test results based on tolerances that are consistent with accident analysis assumptions.

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

The proposed activity does not create the possibility of an accident of a different type than any previously evaluated. No physical plant changes are being made and no new failure modes have been introduced by the proposed changes. This evaluation revises the acceptance criteria for MSSV and PSV lift test results based on tolerances that are consistent with accident analysis assumptions. The actual setpoints and as-left setpoint tolerances of the MSSVs and PSVs are not changed as a result of this evaluation.

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

The margin of safety as defined in the basis for any Technical Specification or in any licensing document has not been reduced. MSSV and PSV setpoint values are not being changed. MSSV and PSV setpoints are still required to be set within a tolerance of plus or minus 1% (the as-left setpoint tolerance). This evaluation allows for the revision of acceptance criteria for MSSV and PSV lift test results such that testing criteria is consistent with accident analysis assumptions. This will allow for the accommodation of setpoint drift without invalidating the accident analyses. The proposed changes are consistent with the Standard Technical Specifications, which require MSSV and PSV setting within a plus or minus 1% tolerance, but allow surveillance testing to accept valves that lift within plus or minus 3%. A review of the plants' accident analyses has identified the plant-specific tolerances that may be used for this surveillance testing.

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

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, et al. (FPL), Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: February 16, 2000.

Description of amendment request: These proposed license amendments will revise the Technical Specifications (TS) to delete references to certain motor operated valve thermal overload protection bypass devices for Unit 2 and to revise the TS for accident monitoring instrumentation for both Units 1 and 2. The proposed amendments also make an administrative change to the Unit 2 TS Index.

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

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

The addition of the new ACTION statements for the Unit 1 accident monitoring instrumentation adds conservatism that does not exist in the current Technical Specifications. These changes are consistent with either FPL's originally proposed license amendment for this instrumentation or consistent with the Technical Specification allowed outage time for the component being monitored (*i.e.,* the auxiliary feedwater pumps). Unit 2 valves MV–21–4A and MV–21–4B were modified to be manually operated valves and no longer perform an accident mitigation function. Unit 2 wide range T_{hot} instrumentation is used to satisfy Regulatory Guide 1.97 accident monitoring requirements.

These Technical Specification changes either correct existing errors or add conservatism to the way the Unit is operated. Based on the above, the physical changes to plant equipment or plant operation would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Accident monitoring instrumentation monitors the process of postulated events, and is not an accident initiator. Unit 2 valves MV–21–4A and MV–21–4B were modified to be manually operated valves and no longer have an active safety function, therefore, these valves are not accident initiators. These Technical Specification changes either correct existing errors or add conservatism to the way the Unit is operated. Based on the above, the physical changes to plant equipment or plant operation would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments do not involve a significant reduction in a margin of safety. FPL determined that these proposed license amendments are necessary to correct existing errors or add conservatism to the way the Unit is operated. As such, the assumptions and conclusions of the accident analyses in the UFSAR [Updated Final Safety Analysis Report] remain valid and the associated safety limits will continue to be met.

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

NRC Section Chief: Richard P. Correia.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 18, 2000.

Description of amendment request: The technical specification (TS) changes are being proposed to provide flexibility of operation. These changes include: (1) The ability to have a standby Safety Injection (SI) pump available during Reactor Coolant System (RCS) reduced inventory conditions with the RCS pressure boundary intact; (2) The ability to respond more rapidly with additional makeup sources than currently established by TSs in the unlikely event of a loss of decay heat removal capability or unexpected reduction in RCS inventory; (3) Realigning a footnote to clarify the allowance of an inoperable SI pump to be energized for testing or filling accumulators; (4) Recognition that a substantial vent area exists for cold overpressure protection when the reactor vessel head is on and the studs are fully detensioned; (5) Limit maneuvering the plant beyond Hot Shutdown when one charging pump is operable; and (6) Establishment of a new value for the open permissive interlock associated with the Residual Heat Removal System suction isolation valves.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The proposed changes do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or manner in which structures, systems, and components perform their intended safety function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (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. Since there are no changes to previous accident analysis, the radiological consequences associated with these analyses remain unchanged; therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not result in a change to the design basis of any plant structure, system, or component. All equipment important to safety will operate as designed. The proposed TS changes in conjunction with administrative controls will provide adequate control measures to ensure component integrity is not challenged. The proposed changes do not cause the initiation of any accident nor create any new failure mechanisms. The changes do not result in any event previously deemed incredible being made credible. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not adversely affect equipment design or operation and there are no changes being made to the TS-required safety limits or safety system settings that would adversely affect plant safety. The proposed TS changes in conjunction with administrative controls will provide adequate control measures to ensure component integrity is not challenged. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 18, 2000.

Description of amendment request: Changes to technical specification (TS) Sections 4.0.5 and 4.4.6.2.2.e are being proposed to clarify that the Inservice Testing (IST) program will be performed in accordance with the requirements of surveillance requirement (SR) 4.0.5 and the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code), instead of Section XI of the ASME Boiler and Pressure Vessel Code.

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 review is presented below:

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

Seabrook Station has proposed to utilize the ASME OM Code-1995 including the 1996 Addenda (OMa Code-1996) for the IST of pumps and valves as an alternative to the requirements of the 1989 Edition of Section XI pursuant to 10 CFR 50.55a(f)(4)(iv) subject to the limitations modifications listed in paragraph (b). The use of the ASME OM Code-1995 including the 1996 Addenda has been evaluated by the NRC (64 FR 51370) and has supplanted Section XI of the 1989 Edition of the ASME Boiler and Pressure Vessel Code as the Code referenced in paragraph (b) for the IST of pumps and valves effective November 22, 1999. The proposed administrative changes only add ASME OM and applicable terms from that Code into the TSs. These proposed changes are administrative in nature and do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The changes to the TSs clarify that the IST program will be performed in accordance with the requirements of SR 4.0.5 and the ASME OM Code and to clarify the surveillance interval requirements for components tested on a Semi-quarterly and Biennial frequency. The proposed changes are administrative in nature and do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The changes to the TSs do not involve a reduction in the margin of safety. As previously identified the subject changes are administrative in nature and will clarify that the IST program will be performed in accordance with the requirements of SR 4.0.5 and the ASME OM Code. The use of the ASME OM Code-1995 including the 1996 Addenda in lieu of Section XI of the ASME Boiler and Pressure Vessel Code will result in a net improvement in the measures for performing the IST of pumps and valves and has been previously evaluated by the NRC. Therefore, the proposed changes to the TSs will not result in a significant reduction in a margin of safety. 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: February 29, 2000:

Description of amendment request: The proposed amendment would approve continued use of two exceptions previously granted by the Nuclear Regulatory Commission (NRC) to the American Society of Mechanical Engineers N510–1989 testing requirements for the emergency filtration train (EFT) system, revise the Technical Specifications (TSs) to reflect modifications to the EFT system that eliminate the need for additional test exceptions, revise the TSs to be consistent with the guidance of NRC Generic Letter 99–02, and revise the TSs to include operability requirements for the EFT system during operations that could result in a fuel handling accident.

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

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

During an accident, the Control Room Emergency Filtration [EFT] System provides filtered air to pressurize the Control Room to minimize the activity, and therefore the radiological dose, inside the Control Room. The SBGT [standby gas treatment] System maintains a small negative pressure in the Reactor Building to minimize ground level escape of airborne radioactivity. Technical Specification operability and surveillance requirements are established in order to ensure that the SBGT and EFT Systems will perform their safety functions during an accident. The proposed amendment documents the test method for laboratory testing of charcoal adsorbers in both systems, implements adequate test acceptance criteria, and improves the methodology of in-place testing of charcoal filters in the EFT System. The additional operability requirements for the EFT System ensure that the systems will be available when required. The surveillances adequately show that the system is operable and capable of performing its safety function. Dose to the public and the Control Room operators are not affected by the proposed change.

Since neither system is an accident initiator, the probability of an accident is not increased.

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

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

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

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes. The proposed surveillance requirements are consistent with industry and regulatory guidance and show that the system is capable of performing its safety function. The added operability requirements for the EFT System ensure that the system will be available when required.

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

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

The proposed amendment is consistent with current industry and regulatory standards for testing filters. The proposed amendment maintains margins of safety. Off-site and Control Room dose assessments are not affected by the proposed amendment, since the ability of the SBGT and EFT Systems to perform their safety function is shown by the proposed surveillance requirements. The proposed change to the surveillances provides assurance that the system will perform at the filter efficiency used in the evaluation of the radiological consequences of the postulated events. Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Claudia M. Craig.

PP&L, Inc., Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: January 13, 2000.

Description of amendment request: The amendment would revise the Technical Specifications (TSs) for both units to clarify Figure 3.4.10–1, ''Reactor Vessel Pressure vs. Minimum Vessel Temperature.'' The amendment would also revise the Unit 2 TS to correct a reference in TS 5.6.5.b, ''Core Operating Limits Report (COLR).''

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

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

This proposal does not involve an increase in the probability or consequences of an accident previously evaluated. The proposed revision to Technical Specification Figure 3.4.10.1 and the proposed revision to the references in the Unit 2 Technical Specification section 5.6.5.b are administrative and/or editorial in nature, and do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revision to Technical Specification Figure 3.4.10.1 and the proposed revision to the references in the Unit 2 Technical Specification section 5.6.5.b are administrative and/or editorial in nature. The proposed revisions do not change any plant systems, structures, or components, nor do they change any existing accident analysis, or create any new or different kind of accident from any accident previously evaluated.

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

This proposal does not involve a significant reduction in the margin of safety. The proposed revision to Technical Specification Figure 3.4.10.1 and the proposed revision to the references in the Unit 2 Technical Specification section 5.6.5.b are administrative and/or editorial in nature, and do not result in [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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PP&L, Inc., 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Marsha Gamberoni, Acting.

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

Date of amendment request: March 8, 2000.

Description of amendment request: The proposed amendment would revise the R. E. Ginna Nuclear Power Plant Improved Technical Specifications associated with the Spent Fuel Pool Storage (SFP) (limiting condition for operation (LCO) 3.7.13), and Design Features Fuel Storage (4.3).

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. Operation of Ginna [Nuclear Power Plant] in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The administrative change only involves how the maximum initial fuel assembly enrichment is described and has no impact on the probability or consequences of an accident. The remaining change is evaluated below.

The regions of the SFP and specific storage cell types differ from each other in regards to the specific absorber material within the cells. Administrative controls are used to maintain the specified storage patterns and to assure storage of a fuel assembly in a proper location based on initial U–235 enrichment, burnup, and decay time. Procedures which perform this surveillance will include independent verification provisions.

There is no significant increase in the probability of an accident concerning the potential insertion of a fuel assembly in an incorrect location in the storage racks. Ginna currently uses administrative controls to move fuel assemblies from location to location within the SFP. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Improved Technical Specification spent fuel rack storage configuration limitations. Fuel movement procedures are planned to include independent verification of fuel handling steps.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks. The criticality safety analysis demonstrate that the pool K_{eff} will remain ≤0.95 following an accidental misloading due to the boron concentration of the pool. The existing Improved Technical Specification limitation on soluble boron within the SFP will ensure that an adequate boron concentration is maintained.

Based on the above, it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously analyzed.

2. Operation of Ginna [Nuclear Power Plant] in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The administrative change to the Improved Technical Specifications has no impact on plant hardware or operations and therefore cannot create a new or different kind of an accident.

Criticality accidents in the SFP are not new or different types of accidents, they have been analyzed in the Updated Final Safety Analysis Report and in criticality safety analysis reports associated with specific licensing amendments for fuel enrichments up to the nominal 5.0 weight percent U–235 that is assumed for the proposed change.

The current Improved Technical Specifications contain limitations on the minimum SFP boron concentration. The proposed changes to the Improved Technical Specifications to allow credit for soluble boron for a K_{eff} < 0.95 in the SFP is consistent with the results of the new criticality safety analysis. Since soluble boron has always been maintained in the SFP water, and is currently required by Improved Technical Specifications, the implementation of this new requirement will have no effect on normal SFP operations and maintenance. A dilution of the spent fuel pool soluble boron has always been a possibility, however, it has been shown in the SFP boron dilution analysis that there are no credible dilution events for which the spent fuel pool K_{eff} could increase to >0.95. Therefore, the implementation of crediting soluble boron in the SFP will not result in the possibility of a new kind of accident.

The proposed changes to Improved Technical Specifications LCO 3.7.13 continue to specify the requirements for the spent fuel rack storage configurations. Since the proposed SFP storage configuration limitations will be similar to the current ones, the new limitations will not have any significant effect on normal spent fuel pool operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will be performed to ensure that the spent fuel pool loading configuration meets specified requirements.

The misloading of a fuel assembly in the required storage configuration has been evaluated. In all cases, the rack K_{eff} remains ≤0.95.

Under the proposed amendment, no changes are being made to the racks themselves, any other systems, or to the physical structures of the Auxiliary Building itself. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna [Nuclear Power Plant] in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed administrative change to the Improved Technical Specifications has no impact on any acceptance criteria, plant operations or the actual failure of any systems, components or structure; therefore the change has no impact on the margin of safety.

The spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality safety analysis performed in a manner analogous to that of the NRC approved Westinghouse spent fuel rack criticality safety analysis methodology.

While the criticality safety analysis utilized credit for soluble boron, storage configurations have been defined using 95/95 K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be <1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the spent fuel pool K_{eff} is maintained at ≤0.95.

The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to K_{eff} exceeding 0.95 has been evaluated and shown to be not credible. An evaluation has been performed which shows that dilution of the SFP boron concentration from 2300 ppm to 975 ppm is not credible. Also, the spent fuel rack K_{eff} will remain <1.0 (with a 95/95 confidence level) with the SFP flooded with unborated water. These analyses demonstrate a level of safety comparable to the conservative criticality safety analysis methodology required by Westinghouse WCAP–14416. Therefore, these changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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, Acting.

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

Date of amendment request: March 15, 2000.

Description of amendment request: The proposed amendment would change to the technical specifications, to provide a completion time of 7 days of continued reactor operation with two CAD subsystems inoperable.

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

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

The safety-related function of the Containment Atmosphere Dilution (CAD) system is to mitigate the effects of a loss-of-coolant-accident (LOCA) by limiting the volumetric concentration of oxygen in the primary containment atmosphere. The CAD System is not an event initiator, therefore, the probability of the occurrence of an accident is not affected by this proposed Technical Specification (TS) change. Emergency procedures preferentially use the normal containment inerting system to provide post-accident vent and purge capability, with the CAD system only serving in a backup role to this system. Hence, in the event of the inoperability of both CAD subsystems, the proposed TS require the normal containment inerting system to be verified available as an alternate oxygen control means.

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

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

This TS change does not result in any changes to the CAD equipment design or capabilities or to the operation of the plant. Since the change impacts only the required action completion time for periods of CAD subsystem inoperability and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

As stated in GL 84–09, a Mark I type boiling water reactor (BWR) plant is not considered to rely upon purge/repressurization systems such as CAD as its primary means of hydrogen control when the unit(s) is operated in accordance with certain technical criteria. The BFN units are operated in accordance with these criteria. The BFN Unit 2 and Unit 3 containments are inerted with nitrogen during normal operation, recycled containment atmosphere is used for pneumatically operated components inside containment, and there are no potential sources of oxygen generation inside containment other than the radiolytic decomposition of water. The system preferred by the EOIs for oxygen control post-accident is the normal primary containment inerting system. Because the probability of an accident involving hydrogen and oxygen production is small, CAD is not the primary system used to mitigate the creation of combustible containment atmosphere mixtures, and because the requested LCO where both CAD subsystems is inoperable is not long, no significant reduction in the margin of safety is associated with this proposed amendment.

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

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

NRC Section Chief: Richard Correia.

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

Date of amendment request: February 18, 2000.

Description of amendment request: The proposed amendment would revise the technical specifications to identify (1) M5 alloy as a material used in the construction of fuel assemblies, and (2) The associated topical report that describes the fuel.

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

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

The proposed TS revision will allow the use of a new advance alloy material for the fuel rod cladding. The new M5 alloy properties are not significantly different than the characteristics of the currently used zircaloy-4 as demonstrated in the NRC approved Topical Report BAW–10227P–A for the use of the M5 alloy for fuel rod cladding. In this topical, the M5 alloy was shown to perform very similar to the zircaloy-4 with improved performance in several areas including fuel cladding corrosion, hydrogen pickup, fuel rod and fuel assembly growth, and fuel rod cladding creep. The proposed revision will not alter the operating characteristics of the plant or plant components. The fuel rod cladding function will not be changed even though some of the rod cladding properties could be enhanced.

The M5 alloy will maintain fuel rod cladding integrity such that the potential for rod cladding failures is not increased. The fuel rod cladding is not assumed to arbitrarily fail as an accident initiator even though it does function to ensure that initial core conditions are within the analysis assumptions and to provide a barrier to the release of radiation. Therefore, the proposed revision will not increase the possibility of an accident based on the new M5 alloy having similar properties as the zircaloy-4 material.

The ability of the new M5 fuel rod cladding material to provide a barrier against the release of radioactive fuel material has not been reduced with respect to the zircaloy-4 material and the generation of hydrogen has been reduced. The approved topical report evaluated postulated accidents that involved adverse core conditions and the release of radionuclides and found the M5 alloy to perform similar to the current fuel rod cladding material. Rod cladding failures are assumed to occur in the fuel handling accident; however, the consequences of this event is independent of the properties of the fuel rod cladding. This is based on the fuel handling event assuming the rupture of fuel rods regardless of the rod cladding material. Therefore, based on the topical report results, the proposed revision to allow the use of M5 fuel rod cladding material will not significantly increase the consequences of an accident and the potential for the release of radioactive material to the environment.

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

The proposed M5 rod cladding material has been demonstrated to have properties that are not significantly different than the current zircaloy-4 in maintaining the integrity of the fuel rods. The new material will not alter the functions of the rod cladding which is to provide a barrier against the release of radioactive material. Initial plant conditions, which is considered in the accident analysis, will also be maintained such that no new plant conditions will exist that could affect the analysis results. Since plant functions and conditions are not impacted by the proposed revision and the new M5 rod cladding is not postulated to become an accident initiator based on the similarity with zircaloy-4, the possibility of a new or different kind of accident is not created.

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

The margin of safety is established by the acceptance criteria used by NRC. Meeting the acceptance criteria assures that the consequences of accidents are within known and acceptable limits. The loss-of-coolant accident (LOCA) acceptance criteria are unchanged: peak cladding temperature of ≤ 2200 degrees Fahrenheit; maximum cladding oxidation of ≤17 percent of the total cladding thickness before oxidation; maximum hydrogen generation of ≤1 percent of the hypothetical amount if all of the cladding metal were to react; coolable geometry such that the core remains amenable to cooling; and long-term cooling to maintain core temperature at an acceptably low value and removal of decay heat for an extended period.

These requirements continue to be met with the new M5 fuel rod cladding material. The acceptance criteria for Departure from Nucleate Boiling (DNB) events has not changed and is still the 95 percent probability and 95 percent confidence interval that DNB is not occurring during the transient. The changes to material properties have been evaluated in BAW–10227P–A and all applicable acceptance criteria are met. In addition, the proposed revision to allow the use of M5 fuel rod cladding will not impact plant setpoints that maintain the margin of safety. Based on these results, it is concluded that the margin of safety is not significantly reduced.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of application request: March 6, 2000 (ULNRC–04197).

Description of amendment request: The proposed amendment will revise Table 3.7.1–1, ''Operable Main Steam Safety Valves [MSSVs] versus Maximum Allowable Power,'' of the technical specifications to reduce the maximum allowable reactor power for a given number of operable MSSVs per steam generator. There are five MSSVs on each of the four steam generators for the plant. This change will increase restrictions on the operation of the plant to account for (1) Westinghouse letter, SCP–99–129, dated July 7, 1999, and (2) Westinghouse Nuclear Safety Advisory Letter, NSAL–94–001, dated January 20, 1994. This change will decrease the setpoint values for the power range neutron flux high channels, which are part of the reactor trip system (RTS) instrumentation in Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' of the TSs, and will result in the reactor being shut down at a lower reactor power for a given number of operable MSSVs per steam generator. There is also a change to the Required Action

A.1 for Limiting Condition for Operation (LCO) 3.7.1, ''Main Steam Safety Valves (MSSVs).'' The licensee has administrative controls in place to ensure that the proposed reduced maximum allowable reactor power values are in effect at the plant.

In addition to the changes to LCO 3.7.1 above, the licensee also proposed to correct two format errors in the actions for LCO 3.7.1. The first correction is to add a separating line between Conditions A and B; the second correction is to move the word ''(continued)'' above the bottom line for Condition B. Neither of these corrections have any affect on the requirements stated in LCO 3.7.1. The licensee also showed the changes to the Bases of LCO 3.7.1 that are related to the proposed amendment including two editorial corrections to the 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operability of the MSSVs ensures that the secondary side system pressure is limited to within 110% of its design pressure during the most severe anticipated system operational transient, which is the Loss of Load/Turbine Trip Event. As stated in FSAR [Callaway final safety analysis report] 15.2.3.3, these events do not present a hazard to the integrity of the reactor core, the reactor coolant system, or the main steam system. The Power Range Neutron Flux High Reactor Trip function and the MSSVs are designed to mitigate the consequences of the Loss of Load/Turbine Trip event. The Loss of Load event is initiated as a result of an electrical system disturbance and the Turbine trip event is initiated as a result of a signal derived from the turbine emergency trip fluid pressure transmitters and turbine stop valve limit switches.

The Power Range Neutron Flux High Reactor Trip function and the MSSVs ensure that the FSAR Loss of Load/Turbine Trip analyses are bounding for cases when not all of the MSSVs are operable. Technical Specification Table 3.7.1–1 controls the Power Range Neutron Flux High Setpoints when a MSSV is found to be inoperable. The controls under this proposed change, which are more restrictive than the ones in Technical Specification Table 3.7.1–1, do not install or modify any plant equipment. The revised Power Range Neutron Flux High Setpoints with inoperable MSSVs proposed under this change are bounded by the reactor trip setpoints currently provided in Table 3.7.1–1. In addition the functionality of plant equipment is unaffected by the proposed change.

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

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

The proposed changes ensure that the FSAR Loss of Load/Turbine Trip analyses are bounding for cases when not all of the MSSVs are operable. Furthermore, the changes do not result in any previously incredible accidents becoming credible. No additional equipment is being [added to the plant or] credited in the mitigation of any [FSAR] Chapter 15 accident events, and the proposed changes do not invalidate any previous conclusions.

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

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

Using the Power Range Neutron Flux High Setpoints with inoperable MSSVs provided by Westinghouse (Reference 2 [in the licensee's application letter]) in lieu of the ones calculated using the equation provided in the Current Technical Specifications Bases, results in more conservative reactor trip setpoints. This increases the margin of safety. The margin of safety as determined in the basis for the Technical Specification is not reduced.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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.

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.

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: February 18, 2000.

Description of amendment request: The amendment requests approval to Baltimore Gas and Electric Company's (BGE's) operating license that the new identified failure mode is acceptable on the basis that BGE will assure on every shift that safety-related loads are sufficiently available to Diesel Generator 1A to ensure the minimum load is met.

Date of publication of individual notice in ***Federal Register:*** March 7, 2000 (65 FR 12038).

Expiration date of individual notice: April 6, 2000.

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

Date of application for amendment: July 26, 1999, as supplemented January 20, 2000.

Brief description of amendment: The proposed amendment would revise Technical Specifications associated with the degraded voltage trip and the under-frequency reactor trip surveillance tests.

Date of publication of individual notice in ***Federal Register:*** February 28, 2000 (65 FR 10565).

Expiration date of individual notice: March 29, 2000.

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 application of amendments: January 27, 2000.

Brief description of amendments: The amendments revised the Facility Operating Licenses by (a) deleting the license conditions that have been fulfilled by actions that have been completed, (b) changing the license conditions that have been superseded by the current plant status, and (c) incorporating other administrative changes.

Date of publication of individual notice in ***Federal Register:*** February 8, 2000 (64 FR 6243).

Expiration date of individual notice: March 9, 2000.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the

Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR chapter I, which are set forth in the license amendment.

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

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

For further details with respect to the action see (1) The applications for amendment, (2) The amendment, and (3) The Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: December 16, 1999.

Brief description of amendment: The amendment allowed a one-time extension of some Technical Specification surveillance intervals to support elimination of a planned spring 2000 midcycle outage. The surveillances would be extended to no later than November 30, 2000.

Date of issuance: March 17, 2000.

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

Amendment No.: 125.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 12, 2000 (65 FR 1921).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2000.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: October 25, 1999 (U–603282).

Brief description of amendment: The amendment revised the Technical Specification allowable values for the reactor protection system electric power monitoring assembly overvoltage and undervoltage trip setpoints.

Date of issuance: March 21, 2000.

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

Amendment No.: 126.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 12, 2000 (65 FR 1919).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 2000.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: May 13, 1999.

Brief description of amendment: The amendment revised Sections 2.a., 2.c.(3) and 2.c.(7) of the Facility Operating License to delete already completed license conditions or update out-of-date reporting references, and made a change to the Bases of Technical Specification 3.1.1 regarding the pressurizer safety valves lift setpoint.

Date of issuance: March 14, 2000.

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

Amendment No.: 222.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications and License.

Date of initial notice in ***Federal Register:*** June 30, 1999 (64 FR 35206).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2000.

No significant hazards consideration comments received: No.

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of application for amendment: November 18, 1999.

Brief description of amendment: The amendment incorporated a change in the pressure-temperature curves in the Calvert Cliffs Nuclear Power Plant, Unit No. 1 Technical Specifications. Baltimore Gas and Electric Company changed the fluence level for which the curves are valid from 2.61×10^{19} n/cm^2 to 4.49×10^{19} n/cm^2.

Date of issuance: March 20, 2000.

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

Amendment No.: 234.

Facility Operating License No. DPR–53: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70078).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 21, 2000.

Brief description of amendment: The amendment changed Technical Specification Surveillance Requirement 4.0.5.f to allow the required examination of weld RH–2005–29 to be deferred until the next scheduled refueling outage or December 31, 2000, whichever is earlier.

Date of issuance: March 22, 2000.

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

Amendment No.: 123.

Facility Operating License No. NPF–18: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (65 FR 11809 dated March 6, 2000). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by April 5, 2000, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final no significant hazards consideration determination are contained in a Safety Evaluation dated March 22, 2000.

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

Date of application for amendment: December 17, 1999, as supplemented January 26, 2000.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement 3.6.1.3.9 to allow a representative sample of reactor instrumentation line excess flow check valves to be tested every 18 months, instead of testing each excess flow check valve every 18 months.

Date of issuance: March 14, 2000.

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

Amendment No.: 137.

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

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4270) The January 26, 2000, letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2000.

No significant hazards consideration comments received: No.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: November 3, 1999, as supplemented by letter dated January 14, 2000.

Brief description of amendments: The amendments revised the Technical Specifications Surveillance Requirements (SR) 3.8.1.13 and SR 3.8.1.14 for emergency diesel generators at Catawba Nuclear Station. Specifically, these SR may now be performed at any operational power level for Catawba Nuclear Station. In addition, in November 3, 1999, application, licensee requested that the power factor requirements be deleted from SR 3.8.1.9, and 3.8.1.14. However, licensee withdrew the power factor deletion part of the request for Catawba Nuclear Station, Units 1 and 2, in a letter dated January 14, 2000.

Date of issuance: March 16, 2000.

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

Amendment Nos.: Unit 1–185; Unit 2–177.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 1, 1999 (64 FR 67332).

The January 14, 2000, letter provided additional clarifications that did not enlarge the scope of the previous no significant hazard consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 16, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 3, 1999, as supplemented by letters dated January 14 and February 17, 2000.

Brief description of amendments: The amendments revise the following Technical Specifications Surveillance Requirements (SR): (1) SR 3.8.1.9 to allow performance of the diesel generator (DG) load rejection test at any operational power level and to delete the power factor requirements, (2) SR3.8.1.10 to allow performance of the DG full load rejection test at any power level, and (3) SR 3.8.1.14 to allow performance of the 24-hr DG run at any operational power level and delete the power factor requirement. No plant modification is involved.

Date of issuance: March 15, 2000.

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

Amendment Nos.: Unit 1–192; Unit 2–173.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 1, 1999 (64 FR 67333).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 2000.

No significant hazards consideration comments received: No.

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 application of amendments: January 27, 2000.

Brief description of amendments: The amendments revised the Facility Operating Licenses by (a) Deleting the license conditions that have been fulfilled by actions that have been completed, (b) Changing the license conditions that have been superseded by the current plant status, and (c) Incorporating other administrative changes.

Date of Issuance: March 13, 2000.

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

Amendment Nos.: Unit 1–311; Unit 2–311; Unit 3–311.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Facility Operating Licenses and License Conditions.

Date of initial notice in ***Federal Register:*** February 8, 2000 (65 FR 6243).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 13, 2000.

No significant hazards consideration comments received: No.

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: October 29, 1999.

Brief description of amendment: This amendment authorizes a revision to the post loss-of-coolant accident (LOCA) dose calculations described in the River Bend Station (RBS) Updated Safety Analysis Report (USAR). The analyses are being updated to account for several changes that were determined by the licensee to involve an unreviewed safety question in accordance with title 10 of the *Code of Federal Regulations,* section 50.59(a)(2)(i). Specifically, the licensee requested the following changes to the RBS USAR, Sections 6.2.3 and 15.6.5:

Increase of the positive pressure period of the secondary containment following a design basis accident to 195.5 seconds from 189 seconds.

Decrease of the suppression pool water volume to 1.2E5 ft^3 from 1.35E5 ft^3 for use in the post-LOCA dose calculation.

Change to the engineered safety feature (ESF) liquid leakage model adding the leakage resulting from a gross failure of a passive component outside of primary containment.

Direct release of ESF leakage through the Standby Gas Treatment System to the environment without hold up in the auxiliary building.

Date of issuance: March 17, 2000.

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

Amendment No.: 111.

Facility Operating License No. NPF–47: The amendment revised the USAR.

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70084).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2000.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: November 18, 1999.

Brief description of amendment: This amendment removes license condition 3.H, ''Long Term Program,'' from Facility Operating License No. DPR–35 for the Pilgrim Nuclear Power Station.

Date of issuance: March 13, 2000.

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

Amendment No.: 183.

Facility Operating License No. DPR–35: Amendment revised the License.

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 6404).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 2000.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: May 5, 1999, as supplemented January 31, 2000.

Brief description of amendment: This amendment modifies the licensing basis for the on-site fuel storage requirements for the emergency diesel generators. Various sections of the technical specifications were amended to reflect the new licensing basis.

Date of issuance: March 17, 2000.

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

Amendment No.: 184.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 2, 1999 (64 FR 29708).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 3, 1998, as supplemented by letter dated October 7, 1999.

Brief description of amendment: This amendment authorizes revision of the Grand Gulf Nuclear Station Updated Final Safety Analysis Report for implementation of a limited scope application of the alternative accident source term described in NUREG–1465. The amendment allows a change in the minimum time assumed for the onset of fission product release from perforated fuel rods following a postulated design basis loss-of-coolant accident.

Date of issuance: March 22, 2000.

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

Amendment No: 143.

Facility Operating License No. NPF–29: The amendment changes the Grand Gulf Nuclear Station design basis by revising the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** December 1, 1999 (64 FR 67333).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 22, 2000.

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 2, 1999.

Brief description of amendment: This amendment revises the Technical Specifications (TSs) to (1) Relocate the requirements of TS 3/4.1.2.8, Reactivity Control Systems—Borated Water Sources—Shutdown, in its entirety, to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM); (2) Relocate the requirements of TS 3/4.1.2.9, Reactivity Control Systems—Borated Water Sources—Operating, to the USAR TRM, except for portions applicable to the Borated Water Storage Tank, which have been deleted because they are redundant to the existing provisions of TS 3/4.5.4, Emergency Core Cooling Systems—Borated Water Storage Tank; (3) Modify TS 3/4.1.2.1, Reactivity Control Systems—Borated Water Sources—Shutdown, by deleting references to TS 3.1.2.8; (4) Incorporate corresponding changes to the TS index; and (5) Incorporate corresponding changes to the TS Bases.

Date of issuance: March 14, 2000.

Effective date: Immediately upon date of issuance and shall be implemented within 120 days.

Amendment No.: 238.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70086).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2000.

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: September 8, 1998.

Brief description of amendment: This amendment revises Technical Specification (TS) 5.3.1, ''Design Features—Reactor Core—Fuel Assemblies,'' and TS Bases Section 2.1, ''Safety Limits.'' The amendment permits the use of the Framatome Cogema Fuels ''M5'' advanced alloy for fuel rod cladding and fuel assembly spacer grids.

Date of issuance: March 15, 2000.

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

Amendment No.: 239.

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

Date of initial notice in ***Federal Register:*** October 7, 1998 (63 FR 53961).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 23, 1999, as supplemented March 9, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) surveillance testing of the safety-related ventilation system charcoal to meet the actions requested in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999. Other systems impacted include the emergency containment filtering system, post accident containment vent system, and the control room emergency ventilation system.

Date of issuance: March 21, 2000.

Effective date: March 21, 2000.

Amendment Nos.: 205 and 199.

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

*Date of initial notice in **Federal Register:*** December 15, 1999 (64 FR 70089). The March 9, 2000, submittal provided clarifying information that did not change the scope of the original request or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 19, 1999, as supplemented February 23, 2000.

Brief description of amendment: Changes the Crystal River Unit 3 Technical Specifications (TS) to incorporate the requirements of 10 CFR 50.55a relating to containment inspections.

Date of issuance: March 16, 2000.

Effective date: March 16, 2000.

Amendment No.: 191.

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

*Date of initial notice in **Federal Register:*** October 20, 1999 (64 FR 56530). The February 23, 2000, supplement did not affect the original no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 16, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 3, 1998.

Brief description of amendments: The amendments incorporate the Distribution Ignition System requirements into the Unit 1 and Unit 2 Technical Specifications.

Date of issuance: March 15, 2000.

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

Amendment Nos.: 242 and 223.

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

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4279).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 2000.

No significant hazards consideration comments received: No

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: March 31, 1999.

Brief description of amendment: The change modifies Cooper Nuclear Station's Technical Specifications, Section 5.3.1, ''Unit Staff Qualifications.'' The change endorses the provisions of Regulatory Guide 1.8, Revision 2, ''Qualification and Training of Personnel for Nuclear Power Plants,'' for the shift supervisor, senior operator, licensed operator, shift technical advisor, and radiological manager.

Date of issuance: March 15, 2000.

Effective date: March 15, 2000, to be implemented within 30 days.

Amendment No.: 181.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 5, 1999 (64 FR 24197).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 23, 1999, as supplemented December 7, 1999.

Brief description of amendment: The amendment updates the list of documents which describe the analytical methods used to determine the core operating limits specified in Technical Specification 6.9.1.8b.

Date of issuance: March 17, 2000.

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

Amendment No.: 242.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4284).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 27, 1999.

Brief description of amendments: Revised the technical specifications to clarify several administrative requirements, delete redundant requirements, and correct typographical errors, and are considered administrative in nature.

Date of issuance: March 14, 2000.

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

Amendment Nos.: Unit 1–139; Unit 2–102.

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

*Date of initial notice in **Federal Register:*** November 17, 1999 (64 FR 62714).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 9, 1996, as supplemented June 6, 1997, and June 7, 1999.

Brief description of amendment: The amendment removes the requirement for the Plant Operating Review Committee review of the fire protection program and implementing procedures.

Date of issuance: March 13, 2000.

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

Amendment No.: 201.

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

*Date of initial notice in **Federal Register:*** December 1, 1999 (64 FR 67339).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 2000.

No significant hazards consideration comments received: No.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: October 24, 1997, as supplemented, January 8, September 21, and December 22, 1998; and January 7, February 17, June 21, and August 23, 1999, and February 7, 2000.

Brief description of amendments: These amendments revise the Salem Technical Specifications (TSs), Section 3/4.7.7, ''Auxiliary Building Exhaust Air Ventilation System,'' to require two auxiliary building ventilation system (ABVS) supply fans, and three ABVS exhaust fans to be operable, and clarify administrative controls.

Date of issuance: March 21, 2000.

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

Amendment Nos.: 228 and 209.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 17, 1997 (62 FR 66140).

The January 8, September 21, and December 22, 1998; and January 7, February 17, June 21 and August 23, 1999; and February 7, 2000, letters provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 2000.

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 or amendments: November 30, 1999.

Brief description of amendments: The amendments revise Technical Specifications and associated Bases to Surveillance Requirement 3.8.1.12 to remove the restriction which prevents performance of the diesel generator 24-hour run while operating in either Mode 1 or 2.

Date of issuance: March 15, 2000.

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

Amendment Nos.: Unit 1–218; Unit 2–159.

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

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73098)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 2000.

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: October 15, 1999.

Brief description of amendments: The amendments revise the Safety Limit Minimum Critical Power Ratios (SLMCPR) in Technical Specification 2.1.1.2 to reflect the results of a cycle-specific calculations for Unit 1 Cycle 19 and Unit 2 Cycle 16. The calculations were performed using the new NRC-approved methodology for determining SLMCPRs.

Date of issuance: March 22, 2000.

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

Amendment Nos.: Unit 1–219; Unit 2–160.

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

*Date of initial notice in **Federal Register:*** November 17, 1999 (64 FR 62715 and 64 FR 62716).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 22, 2000.

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: April 6, 1999.

Brief description of amendments: The amendments revise the Technical Specifications to allow an increase of 168 fuel assemblies in the storage capacity of Unit 1's spent fuel pool and an increase of 88 fuel assemblies in the storage capacity of Unit 2's spent fuel pool.

Date of issuance: March 23, 2000.

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

Amendment Nos.: Unit 1–220; Unit 2–161.

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

*Date of initial notice in **Federal Register:*** May 4, 1999 (64 FR 23877).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 23, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 12, 1999 (PCN–505).

Brief description of amendments: The amendments revise Technical Specification 5.5.2.13, ''Diesel Fuel Oil Testing Program.''

Date of issuance: March 20, 2000.

Effective date: March 20, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—167; Unit 3–158.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 1, 1999 (64 FR 67339).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 28, 1999, as supplemented February 4, 2000 (TS–399).

Brief description of amendments: The Technical Specifications (TS) have been changed to increase the allowable leakage for any one of the four main steam line (MSL) penetrations from 11½ to 100 standard cubic feet per hour (scfh), and to establish a 150 scfh limit on the maximum allowable combined leakage of all four MSL penetrations.

Date of issuance: March 14, 2000.

Effective date: March 14, 2000.

Amendment Nos.: 263 and 223.

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

*Date of initial notice in **Federal Register:*** November 3, 1999 (64 FR 59807). The supplemental letter dated February 4, 2000, contained clarifying information that did not change the initial no significant hazards determination.

The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated February 22, 2000, and a Safety Evaluation dated March 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 30, 1999, as supplemented February 29, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TS) analytical methods for core operating limits to implement an analysis supporting a more negative moderator temperature coefficient for the end-of-cycle, rated thermal power condition.

Date of issuance: March 14, 2000.

Effective date: March 14, 2000.

Amendment No.: 20.

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

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4291). The supplemental letter dated February 29, 2000, contained clarifying information and did not change the initial proposed No Significant Hazards Consideration Determination or expand the application beyond the scope of the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 25, 1999, as supplemented January 25, 2000.

Brief description of amendment: The amendment changes the Technical Specifications (TS) to apply the Westinghouse generic best estimate large break loss-of-coolant accident analysis methodology, using the WCOBRA/TRAC code to the Watts Bar Unit 1 plant.

Date of issuance: March 17, 2000.

Effective date: March 17, 2000.

Amendment No.: 21.

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

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 611). The January 25, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 15, 1999 (TS 99–16).

Brief description of amendment: The amendment changes the methodology and frequency for sampling the ice condenser ice bed (stored ice) and adds a new Technical Specification (TS) and associated Bases to change the methodology and frequency for sampling requirements for all ice additions to the ice bed.

Date of issuance: March 21, 2000.

Effective date: March 21, 2000.

Amendment No.: 22.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70092).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 28, 1999, as supplemented December 21, 1999.

Brief description of amendments: The amendments remove the operability and surveillance requirements of Technical Specifications (TS) Section 3/4.6.4.3, ''Waste Gas Charcoal Filter System,'' from the TS and relocate them to the Technical Requirements Manual.

Date of issuance: March 13, 2000.

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

Amendment Nos.: 222 and 203.

Facility Operating License Nos. NPF–4 and NPF–7. Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 6412).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 13, 2000.

No significant hazards consideration comments received: No.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: July 1, 1999.

Brief description of amendments: These amendments reflect a change to Technical Specification Section 15.5.4. The amendments remove one of the two separate methods for verifying the acceptability of reactor fuel for placement and storage in the spent fuel pool and new fuel storage vault.

Date of issuance: March 20, 2000.

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

Amendment Nos.: 194 and 199.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 28, 1999 (64 FR 40911).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 2000.

No significant hazards consideration comments received: No.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: November 15, 1999.

Brief description of amendments: This amendment changes the control rod surveillance interval in TS Table 15.4.1–2, Item 10, ''Partial movement of all rods,'' from once ''Every 2 weeks'' to ''Quarterly.'' This change implements the recommendation of NRC Generic Letter 93–05, ''Line Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

Date of issuance: March 22, 2000.

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

Amendment Nos.: 195 and 200.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73103).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 22, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 22, 1999, as supplemented on December 2, 1999, and January 17, 2000.

Brief description of amendment: The amendment extends the application of the length-based pressure boundary definition (L-criterion) for the Westinghouse mechanical hybrid expansion joints in sleeved steam generator tubes to the end of operating cycle 24.

Date of issuance: March 15, 2000.

Effective date: Immediately upon its date of issuance and is to be implemented within 30 days of the date of issuance.

Amendment No.: 146.

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

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4266).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 2000.

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: October 21, 1999.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.4.10, Pressurizer Safety Valves [PSV], of the improved Technical Specifications (TSs) issued March 31, 1999. The amendment reduced the safety valve set pressure in Limiting Condition for Operation (LCO) 3.4.10 and decreased the setpoint in Surveillance Requirement (SR) 3.4.10.1. The PSV setpoint and setpoint tolerance were changed from 2485 psig ±1% to 2460 psig ±2% in the LCO. The tolerance of ±1% in the SR for resetting the setpoint after testing, it needed, was not changed.

Date of issuance: March 23, 2000.

Effective date: March 23, 2000, and shall be implemented before the restart from refueling outage 11, which is the next refueling outage scheduled to begin October 2000.

Amendment No.: 133.

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

*Date of initial notice in **Federal Register:*** November 17, 1999 (64 FR 62718).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 23, 2000.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant 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, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 5, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. 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 20555–0001, Attention: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

Date of amendment request: February 18, 2000, as supplemented March 8, 2000.

Description of amendment request: The amendment changes current Technical Specification (TS) 4.9a.2 and improved TS 3.7.5 and its associated bases to remove requirements associated with the backup steam supply to turbine-driven auxiliary feedwater pump P–8B.

Date of issuance: March 14, 2000.

Effective date: As of the date of issuance and shall be implemented within 30 days, except that implementation with respect to the improved TSs shall be on or before October 31, 2000.

Amendment No. 190.

Facility Operating License No. DPR–20: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (65 FR 11089, March 1, 2000). 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 31, 2000, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 14, 2000.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of application for amendment: February 25, 2000.

Brief description of amendment: The amendment revised Technical Specification Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation'' to provide a one-time exception, until the next time the turbine is removed from service, from the requirement to perform response time testing for the solenoid valve 1–FSV–47–027. The amendment also supersedes the Notice of

Enforcement Discretion granted on February 23, 2000, and confirmed by letter dated February 25, 2000 (00–6–004).

Date of issuance: March 22, 2000.

Effective date: March 22, 2000.

Amendment No.: 23.

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

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (65 FR 11348 dated March 2, 2000). 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 15, 2000, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated March 22, 2000.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Dated at Rockville, Maryland, this 29th day of March 2000.

For The Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–8211 Filed 4–4–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–286]

Power Authority of the State of New York; Facility Operating License No. DPR–64, Receipt of Petition for Director's Decision Under 10 CFR 2.206

Notice is hereby given that by Petition dated February 10, 2000, Mr. David A. Lochbaum, on behalf of the Union of Concerned Scientists (Petitioner), has requested that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to the Indian Point Nuclear Generating Unit No. 3 (IP3), owned and operated by the Power Authority of the State of New York (the licensee). The Petitioner requests that the NRC order the licensee to assess the corrective action program and the work environment at IP3 and to take immediate actions to remedy any deficiencies they identify. The Petitioner requested that this order be closed out before the sale of IP3 is authorized.

As the basis for this request, the Petitioner states that the NRC's new safety monitoring program assumes that the licensee has both a safety-conscious work environment and an effective method of correcting identified problems. In support of this request, the Petitioner cites concerns by a former member of the licensee's Operations Review Group (ORG) that the corrective action process at IP3 is not effective and that the work environment in the ORG is not safety-conscious. The Petitioner also cites several NRC letters that point out deficiencies in the licensee's corrective action program and one letter that points out an apparent instance of discrimination against an employee who raised safety concerns. In a telephone conference on February 16, 2000, the Petitioner voiced concern that under the NRC's new risk-informed inspection process a breakdown in the licensee's corrective action procedures for a non safety-related system would not be pursued. The Petitioner expressed concern that NRC inspectors might not be able to identify a programmatic breakdown in the licensee's corrective action process before such a breakdown affected plant safety.

The request is being treated pursuant to 10 CFR 2.206 of the Commission's regulations. The request has been referred to the Director of the Office of Nuclear Reactor Regulation. As provided by Section 2.206, appropriate action will be taken on this Petition within a reasonable time.

A copy of the Petition is available for inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 24th day of March 2000.

For the Nuclear Regulatory Commission.

Jon R. Johnson,

Acting Director, Office of Nuclear Reactor Regulation.

[FR Doc. 00–8335 Filed 4–4–00; 8:45 am]

BILLING CODE 7590–01–U

POSTAL SERVICE

Request for Comments on Revising and Updating Five-Year Strategic Plan, Pursuant to the Government Performance and Results Act of 1993

AGENCY: Postal Service.

ACTION: Request for comments.

SUMMARY: The Government Performance and Results Act of 1993 (GPRA) mandated, in 1997, that the Postal Service publish a five-year plan outlining its goals, targets, and strategies, and that the Postal Service update and revise its five-year plan at intervals of no less than three years. In so doing, GPRA states that the Postal Service must, as an aspect of its strategic planning process, solicit and consider the ideas, knowledge, and opinions of those potentially affected by or interested in its Five-Year Strategic Plan. This notice, therefore, asks for public comment concerning the development and drafting of the Postal Service's Five-Year Strategic Plan for fiscal years 2001–2005.

DATES: Comments must be received by May 15, 2000.

ADDRESSES: Written comments should be directed to Robert A.F. Reisner, Vice President, Strategic Planning, United States Postal Service, 475 L'Enfant Plaza SW, Washington, DC 20260–1520. Comments may also be sent to: stratpln@email.usps.gov.

FOR FURTHER INFORMATION CONTACT: Paul Van Coverden, (202) 268–8130.

SUPPLEMENTARY INFORMATION:

Statutory Background

The Government Performance and Results Act of 1993, Pub. L. 103–62 (GPRA), was enacted to make federal programs more effective and publicly accountable by requiring agencies to institute results-driven improvement efforts, service-quality metrics, and customer satisfaction programs. Other statutory goals were to improve Congressional decision making and the internal management of the United States Government, as cited in Pub. L. 103–62, sec. 2(b), 107 Stat. 285. Because of the Postal Service's role as an independent establishment of the Executive Branch of the Government of the United States, section 7 of the law establishes separate provisions which apply to the Postal Service (sections 2801–2805 of title 39, United States Code).

Section 2802 of title 39, United States Code, required that the Postal Service submit to the President and the Congress a strategic plan for its program