8065) between 7:30 a.m. and 4:30 p.m. (EDT). Persons planning to attend this meeting are urged to contact the 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: May 22, 2001.

James E. Lyons,

Associate Director for Technical Support.

[FR Doc. 01–13488 Filed 5–29–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of May 28, June 4, 11, 18, 25, July 2, 2001.

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

STATUS: Public and Closed.

Matters To Be Considered

Week of May 28, 2001

Wednesday, May 30, 2001
10:25 a.m.—Affirmation Session (Public Meeting) (If needed)

Week of June 4, 2001—Tentative

Tuesday, June 5, 2001
9:25 a.m.—Affirmation Session (Public Meeting) (If needed)
2:00 p.m.—Discussion of Management Issues (Closed-Ex. 2)
Wednesday, June 6, 2001
10:30 a.m.—All Employees Meeting (Public Meeting)
1:30 p.m.—All Employees Meeting (Public Meeting)

Week of June 11, 2001—Tentative

Thursday, June 14, 2001
9:55 a.m.—Affirmation Session (Public Meeting) (If needed)
10:00 a.m.—Meeting with Nuclear Waste Technical Review Board (Public Meeting)
1:30 p.m.—Briefing on License Renewal Program (Public Meeting) (Contact: David Solorio, 301–415–1973)

Week of June 18, 2001—Tentative

There are no meetings scheduled for the Week of June 18, 2001

Week of June 25, 2001—Tentative

Wednesday, June 27, 2001
9:25 a.m.—Affirmation Session (Public Meeting) (If needed)

Week of July 2, 2001—Tentative

There are no meetings scheduled for the Week of July 2, 2001

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: David Louis Gamberoni (301) 415–1651.

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

http://www.nrc.gov/SECY/smji/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 the distribution, please contact the Office of the Secretary, Washington, D.C. 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving the Commission meeting schedule electronically, please send an electronic message to dkw@nrc.gov.

Dated: May 24, 2001.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 01–13605 Filed 5–25–01; 10:16 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 May 7, 2001 through May 18, 2001. The last biweekly notice was published on May 16, 2001 (66 FR 27174).

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

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

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

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

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

By June 29, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

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

Date of amendments request: May 1, 2001.

Description of amendments request: The proposed amendments would revise the pressure-temperature limits curves contained in Technical Specification 3.4.9, ''RCS Pressure and Temperature (P/T) Limits.''

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

1. The Proposed License Amendments Do Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The changes to the calculation methodology for the pressure-temperature limits are based on American Society of Mechanical Engineers (ASME) Code Case N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, Division 1,'' and provide adequate margin in the prevention of a non-ductile type fracture of the reactor pressure vessel. The code case was developed based upon the knowledge gained through years of industry experience. The pressure-temperature limits developed using the allowances of ASME Code Case N–640 provide more operating margin. However, experience gained in the areas of fracture toughness of materials and pre-existing undetected defects shows that some of the existing assumptions used for the calculation of pressure-temperature limits are unnecessarily conservative and unrealistic. Therefore, use of the allowances of ASME Code Case N–640 in developing the pressure-temperature limits will provide adequate protection against nonductile-type fractures of the reactor pressure vessel.

Development of the revised BSEP [Brunswick Steam Electric Plant], Unit 1 and 2 pressure-temperature limits was performed using the approved methodologies of 10 CFR 50, Appendix G, and using the allowances of ASME Code Case N–640. The pressure-temperature limits generated using these methods ensure the pressure-temperature limits will not be exceeded during any phase of reactor operation. Therefore, the probability of occurrence and the consequences of a previously analyzed event are not significantly increased. Finally, the proposed changes will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events.

Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes to the pressure-temperature limits.

2. The Proposed License Amendments Will Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

The proposed changes provide more operating margin in the pressure-temperature limits for hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with the benefits being primarily realizable during the pressure tests. The changes also extend the pressure-temperature limits for use up to 32 EFPY [effective full-power years] of operation. However, operation in the ''new'' regions of the pressure-temperature limits has been analyzed and will provide adequate protection against a nonductile-type fracture of the reactor pressure vessel. Otherwise, the proposed pressure-temperature limits do not result in any new or unanalyzed operation of any system or piece of equipment important to safety and, as a result, the possibility of a new type event is not created.

3. The Proposed License Amendments Do Not Involve a Significant Reduction in a Margin of Safety

The revised pressure-temperature limits provide more operating margin and operational flexibility than the existing pressure-temperature limits. With the increased operational margin, a reduction in the safety margin results with respect to the existing limits. However, the industry experience since the inception of pressure-temperature limits confirms that some of the existing methodologies used to develop pressure-temperature limits are unrealistic and unnecessarily conservative. Accordingly, ASME Code Case N–640 takes advantage of this acquired knowledge by establishing more realistic methodologies for the development of pressure-temperature limits. Therefore, operational flexibility is gained and an acceptable margin of safety to reactor pressure vessel non-ductile type fracture is maintained. Evaluation of the revised pressure-temperature limits for use up to 32 EFPY was performed using 10 CFR 50 and ASME Code Case N–640; thus, the margin of safety is not significantly reduced as the result of the proposed changes.

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

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

NRC Section Chief: Patrick M. Madden, Acting.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: April 26, 2001.

Description of amendment request: The proposed amendment would add Section 6.22, ''Reactor Coolant Pump Flywheel Inspection Program'' to Section 6, ''Administrative Controls'' of the Technical Specifications (TSs) and relocate the requirements of TS 3/4.4.10, ''Reactor Coolant System, Structural Integrity'' to the Millstone Unit No. 2 Technical Requirements Manual (TRM). The Bases of the affected TSs would also be relocated to the TRM. The Index pages would also be updated to reflect these changes.

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

1. Involve a Significant Increase in the Probability or Consequences of An Accident Previously Evaluated

Missile generation from a Reactor Coolant Pump (RCP) flywheel could damage the Reactor Coolant System, the Containment, or other equipment or systems important to safety. The fracture mechanics analyses conducted to support the change to Inservice Inspection (ISI) requirements in accordance with the proposed Section 6.22, ''Reactor Coolant Pump Flywheel Inspection Program'' shows that a pre-existing crack sized just below the detection level will not grow to the flaw size necessary to create flywheel missiles within the life of the plant. This analysis conservatively assumes minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area, and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in a Millstone Unit No. 2 flywheel will not grow to the allowable flaw size under Loss of Coolant Accident (LOCA) conditions over the life of the plant, reducing the ISI requirements for the detection of such cracks over the life of the plant will not significantly increase the probability or consequences of an accident previously evaluated

The proposed Technical Specification changes to relocate the requirements for Technical Specification 3/4.4.10, ''Reactor Coolant System, Structural Integrity'' (with the exception of the RCP inspection requirements) to the TRM will have no adverse effect on plant operation or the availability or operation of any accident mitigation equipment. Therefore, the Reactor Coolant System structural integrity (with the exception of the RCP flywheel which is addressed above) will not adversely impact an accident initiator and can not cause an accident. Therefore these changes will not increase the probability or consequences of an accident previously evaluated.

The Index pages will be updated to reflect the proposed changes. These changes are administrative in nature. These changes will not increase 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 will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a Significant Reduction in a Margin of Safety

The fracture mechanics analyses conducted to support the change to ISI requirements in accordance with the proposed Section 6.22, ''Reactor Coolant Pump Flywheel Inspection Program'' shows that significant conservatism has been used for calculating the allowable flaw size, critical flaw size, and crack growth rate in the RCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in a Millstone Unit No. 2 flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under LOCA conditions over the life of the plant, reducing ISI requirements for the detection of such cracks over the life of the plant will not involve a significant reduction in the margin of safety. The proposed changes have no impact on plant equipment operation. Therefore, the proposed changes will not result in a reduction in a margin of safety.

Relocation of Technical Specification 3/4.4.10 (whole specification except the portion specifying surveillance requirement for the RCP flywheel) to the TRM does not imply any reduction in its importance in ensuring that the structural integrity and operational readiness of ASME Code Class 1, 2, and 3 components will be maintained at an acceptable level throughout the life of the plant. The proposed change has no impact on plant equipment operation. Therefore, the proposed change will not result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2 (ANO–1&2), Pope County, Arkansas

Date of amendment request: January 27, 2000, as supplemented by letter dated March 1, 2001.

Description of amendment request: The proposed changes to Arkansas Nuclear One, Unit 1 (ANO–1) and Unit 2 (ANO–2), Technical Specifications (TSs) allow for the qualified condensate storage tank (QCST) to be used for both units as the preferred source of water for emergency feedwater (EFW). Currently, the QCST is aligned to the ANO–1 EFW system while ANO–2 relies on non-safety related tanks and an automatic switchover to the Service Water System as the source of EFW coolant water.

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 staff's analysis is presented below.

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The condensate storage tanks provide a source of condensate grade water for the EFW System. The tanks, one for ANO–1 and two for ANO–2, are already included in the plant TSs. The proposed change allows for both units to operate while aligned to the QCST, but does not affect the physical design, construction, or operation of the condensate storage tanks. These tanks are not associated with the precursors of any accident. This change does not increase the probability of any accident previously evaluated.

As a source of EFW, the tanks serve an accident mitigation function. The proposed change does not alter this function. In addition to the tanks, the Service Water System is also available as a long-term assured source of EFW. The proposed change allows the use of the QCST as the preferred source of EFW for both units. The combination of available sources of water for EFW assures that both units are able to respond to accidents previously evaluated. Because this function continues to be assured, the proposed changes do not increase the consequences of a previously evaluated 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 Accident Previously Evaluated

The condensate storage tanks serve an accident mitigation function as a temporary source of EFW. These tanks have not been identified as a precursor to any accident previously evaluated. The design and operation of these tanks have not changed. While the proposed change does permit the qualified tank to be used by ANO–2, the design has been evaluated and it has been demonstrated that the existing tank is capable of meeting the intended design function of both units.

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

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

The existing sources of water for the ANO–1 EFW system will continue to ensure adequate EFW system performance after the proposed change. The QCST and, if necessary, the Service Water System will ensure that the EFW system performs to maintain margins of safety. The Service Water System is the assured long-term source of cooling water for both units. The safety function of decay heat removal and core cooling continues to be met. There is no reduction in the margin of safety for ANO–1.

The proposed change to the ANO–2 specifications will provide a qualified alternative source of EFW. The required function of the tanks is the same as for ANO–1; that is, to provide a source of water until the unit can successfully transfer to decay heat cooling or until the Service Water System is aligned for long-term cooling. The addition of this QCST to the specification as an alternative to the existing tanks does not decrease the margin of safety.

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

Based on this analysis, 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: May 2, 2001.

Description of amendment request: The proposed amendment would relocate the requirements for the containment recirculation system from the technical specifications to the Technical Requirements Manual (TRM).

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

1. Will Operation of the Facility in Accordance With This Proposed Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

The containment recirculation fans, along with the containment cooling units and containment spray systems, provide a means of circulating the containment atmosphere to ensure adequate mixing of the containment atmosphere. The containment cooling units and containment spray systems are safety-related systems and required by TS 3.6.2.3 and 3.6.2.1, respectively. Adequate air mixing is assured with the use of these two systems. The containment recirculation fans are not credited in the mitigation of any accidents.

Based on an evaluation of the criteria listed in 10 CFR 50.36(c)(2)(ii), the relocation of the containment recirculation fans to the TRM is acceptable.

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

2. Will Operation of the Facility in Accordance With This Proposed Change Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

The containment recirculation fans are not accident initiators. The function they fulfill will continue to be maintained by the containment cooling units and containment spray pumps. Because the proposed amendment will not change the design, configuration or method of plant operation, it will not create the possibility of a new or different kind of accident.

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

3. Will Operation of the Facility in Accordance With This Proposed Change Involve a Significant Reduction in a Margin of Safety?

Air mixing of the containment atmosphere can be accomplished following a LOCA [loss-of-coolant accident] by the containment recirculation fans, the containment cooling units, or the containment spray systems. Any one of these systems is capable of providing adequate air mixing. The proposed change does not change the design function of the containment recirculation fans. Additionally, the containment recirculation fans are not credited in any accident analysis. Since adequate mixing of the containment atmosphere is credited through the containment cooling units and spray systems, relocation of the containment recirculation fan requirements to the TRM does not result in any impact to the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 3, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island Nuclear Station] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271), on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated May 3, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 [Three Mile Island Nuclear Station, Unit 2] accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, PSEG Nuclear LLC, and Atlantic City Electric Company, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: February 8, 2001.

Description of amendment request: The proposed amendment would revise the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, technical specifications (TSs) and the associated TS Bases, to reflect changes to support the activation of the trip outputs of the oscillation power range monitor (OPRM) portion of the power range neutron monitoring (PRNM) system and delete the interim corrective action requirements from the TSs. The OPRM trip function provides protection from exceeding the fuel minimum critical power ratio (MCPR) safety limit in the event of thermal-hydraulic power oscillations. PBAPS is currently operating under interim corrective actions that specify restrictions on plant operations and actions by operators in response to power oscillations. The OPRM system provides an automatic reactor trip which eases the burden on the operators if power oscillations were to occur.

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

This modification has no impact on any of the existing PRNM functions. It connects the OPRM trip function to the reactor protection system; connects the associated trip alarm to the annunciator circuitry; updates the TSs to add the OPRM-related functions and to delete Interim Corrective Actions (ICAs) related requirements; and revises affected procedures.

The ICAs include a restricted region on the power-to-flow map where thermal-hydraulic instabilities were known to be more likely. Operation in the restricted region requires more frequent monitoring of the average power range monitors (APRMs) and local power range monitors (LPRMs), which are part of the PRNM system. This restricted region is less than 10 percent of the full power-to-flow map. Plant operation in portions of the former restricted region without the increased monitoring of APRMs and LPRMs previously required by the ICAs may cause a slight increase in the probability of occurrence of an instability. This potential increase in probability is not significant because operation in this region will still result in a low likelihood of core power oscillations. Because of the more reliable detection of an instability event, should it occur, the automatic scram if preset limits are exceeded, and the elimination of dependence on the operator, the consequences of an instability event are not increased with this modification.

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

2. The Proposed Amendment Does Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

Enabling the OPRM reactor scram function does not create any new system interactions except for the reactor scram function. The failure modes for the new OPRM circuits would be to initiate a reactor scram unnecessarily, or to fail to initiate a reactor scram when instabilities were present. These failures would not create the possibility of a new or different kind of accident. Since the present system has no automatic reactor scram for instabilities, the operators insert a manual scram if necessary, and the effect of core instabilities has been analyzed. The use of a manual scram is still available with the OPRM scram function enabled. Removing the ICAs from the TSs does not create the possibility of a new or different kind of accident, since the effect of core instabilities has been evaluated.

Based on the above discussion, 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 Does Not Involve a Significant Reduction in a Margin of Safety.

The current safety analyses assume that the existing ICA related TS requirements are adequate to prevent exceeding the MCPR safety limit due to an instability event. As a result, there is currently no quantitative or qualitative assessment of an instability event with respect to its impact on MCPR.

The OPRM trip function is being implemented to automate the detection (via direct measurement of neutron flux) and subsequent suppression (via reactor scram) of an instability event prior to exceeding the MCPR safety limit. The OPRM trip provides a trip output of the same type as currently used for the APRMs. Its failure modes and types are identical to those for the present APRM output. Currently, the MCPR safety limit is not impacted by an instability event since the event is mitigated by manual means via the ICAs. In both methods of mitigation (manual and automated), the margin of safety associated with the MCPR safety limit is maintained.

Therefore, based on the fact that the MCPR safety limit will not be exceeded as a result of an instability event following implementation of the OPRM trip function in place of the existing manual ICAs, it is concluded that the proposed amendment does not reduce a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Shippingport, Pennsylvania

Date of amendment request: March 28, 2001.

Description of amendment request: The requested amendment proposes changes to the BVPS–1 Technical Specifications (TSs) associated with the reductions of the reactor coolant system and secondary coolant system specific activity limits. These TS changes support a revised main steam line break safety analysis with a higher assumed primary-to-secondary leak rate in accordance with the methodology described in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes by Outside Diameter Stress Corrosion Cracking.'' TS Bases and other administrative changes are proposed for consistency.

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. Does the Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

The proposed change involves the reduction of the Technical Specification Dose Equivalent Iodine 131 (I–131) activity limits for the reactor coolant system (RCS) and the secondary system which facilitates an increase in the assumed accident-induced primary-to-secondary leak rate in the event of a postulated main steam line break (MSLB) accident. There are no proposed changes to any facility structures, systems, or components. The proposed changes do not affect any initiators of accidents previously evaluated nor does the proposed change introduce any new failure mechanisms that may initiate a previously-evaluated accident. Furthermore, the proposed change would not affect the ability of any accident mitigation system to perform its design-basis function as defined in the Updated Final Safety Analysis Report (UFSAR). The dose consequence analysis for a postulated MSLB accident are being revised as part of this amendment and the resulting calculated dose consequences do not increase.

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

2. Does the Change Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

The proposed change is only associated with the reduction of the RCS and secondary system I–131 activity limits and does not involve changes to any facility structures, systems, or components. There are no proposed changes to the facility or its operation. Since there are no changes being made to any structures, systems, or components, no new failure mechanisms are introduced by the proposed changes that would result in the occurrence of a new or different kind of accident from any accident previously evaluated. The accident analyses contained in the UFSAR continue to remain bounding with regard to the spectrum of possible accidents.

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

3. Does the Change Involve a Significant Reduction in a Margin of Safety?

There are no proposed changes to any structure, systems, or components. Changes proposed to the dose consequence analysis for a postulated MSLB accident are included in the amendment request. The reduction in the RCS and secondary system activity limits are being made to offset the effects of an increased accident-induced primary-to-secondary leak rate resulting from a postulated MSLB accident in accordance with GL 95–05. The margins to safety that could be affected are those associated with the resulting calculated doses to the public and facility personnel. However, the dose-decreasing effect of lowering the activity limits offsets the dose-increasing effect of raising the assumed accident-induced primary-to-secondary leak rate. Consequently, the resulting calculated doses do not increase.

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

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia (Acting).

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

Date of amendment request: March 30, 2001.

Description of amendment request: The proposed amendment would involve changes to Technical Specification (TS) 3/4.5.2, Emergency Core Cooling—ECCS Subsystems—T_{avg} ≥ 280°F.

Technical Specification Limiting Condition for Operation (LCO) 3.5.2 requires two independent Emergency Core Cooling Systems (ECCS) Subsystems to be operable. Surveillance Requirement (SR) 4.5.2.f requires each ECCS subsystem to be demonstrated operable by performing a vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System valves DH–11 and DH–12 that assures the motor operator on valves DH–11 and DH–12 will not be flooded for at least (7) days following a Loss-of-Coolant Accident (LOCA). The test is required to be performed: (1) At least once per 18 months, (2) After each opening of the watertight enclosure, and (3) After any maintenance on or modification to the watertight enclosure which could affect its integrity.

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. Initial conditions and assumptions remain as previously analyzed for accidents in the Davis-Besse Nuclear Power Station Updated Safety Analysis Report.

The proposed changes would increase the surveillance test interval in Technical Specification 4.5.2.f.1 from 18 to 24 months for the vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System valves DH–11 and DH–12. The surveillance data and maintenance records have been reviewed and support an increase in the surveillance test interval from 18 to 24 months based on the low potential for a significant increase in the failure rate of the watertight enclosure due to an increased surveillance interval, and based on the introduction of no new failure modes. The proposed change to the surveillance interval has been evaluated consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' dated April 2, 1991. The watertight enclosure and its condition do not contribute to the initiation of any accident.

Therefore, the probability of any accident previously evaluated is not increased.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the integrity of the watertight enclosure sealing mechanisms has been evaluated, and it has been determined that the sealing mechanisms will remain intact for the proposed increased surveillance interval. Therefore, there is assurance that the backup boric acid precipitation control flow path will remain available, so that there will be no impact on the source term, containment isolation or radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not alter the manner in which the watertight enclosure is sealed or tested, and the operability requirements of Decay Heat Removal System valves DH–11 and DH–12 will continue to be adequately addressed by Surveillance Requirement 4.5.2.f.1.

No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval. An increase in the surveillance test interval from 18 to 24 months is justified based on the low potential for a significant increase in the failure rate of the watertight enclosure due to an increased surveillance interval, and based on the introduction of no new failure modes.

No different accident initiators or failure mechanisms are introduced by the proposed change. Thus, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Not involve a significant reduction in a margin of safety.

An increase in the surveillance test interval from 18 to 24 months is justified based on the low potential for a significant increase in the failure rate of the watertight enclosure due to an increased surveillance interval, and based on the introduction of no new failure modes.

Since there are no new or significant changes to the initial conditions contributing to accident severity or consequences, there are no significant reductions in a margin of safety.

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 1, 2001.

Description of amendment request: The proposed amendment would add new Technical Specification (TS) Administrative Controls Section 6.17, TS Bases Control Program, and make a related change to the TS Index. The proposed new TS Administrative Control would provide requirements for changing and updating the TS Bases. This proposed new TS is similar to the Specification 5.5.14 of NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 1, April 1995.

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. The amendment application proposes to add a new Technical Specification (TS) Administrative Controls Section 6.17, ''Technical Specifications (TS) Bases Control Program,'' and to make a related change to the TS Index. The proposed changes do not involve a change to any structure, system, or component or to the assumptions of any accident analyses.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no equipment, accident conditions, or assumptions are affected which could lead to a significant increase in radiological consequences.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new or different accident initiators are introduced by these proposed changes.

3. Not involve a significant reduction in a margin of safety because there are no new or significant changes to the initial conditions contributing to accident severity or consequences. Consequently, there are no significant reductions in a margin of safety.

On the basis of the above, the DBNPS has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 1, 2001.

Description of amendment request: The proposed amendment would involve changes to Technical Specification (TS) 3/4.3.1, Reactor Protection System Instrumentation; 3/4.3.2.1, Safety Features Actuation System Instrumentation; TS 3/4.3.2.2, Steam and Feedwater Rupture Control System Instrumentation; and Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation.

The proposed changes would revise TS Table 3.3–3, Safety Features Actuation System (SFAS) Instrumentation, TS Table 3.3–11, Steam and Feedwater Rupture Control System (SFRCS) Instrumentation, and associated Bases to add a provision to allow an eight-hour delay in entering an Action statement when an SFAS or SFRCS instrumentation channel is undergoing Channel Functional Testing. The proposed changes would provide a reasonable time to perform the required surveillance testing and relieve the control room staff of the burden of making multiple Action statement entries and exits in order to complete the testing. Additionally, Surveillance Requirements 4.3.1.1.2, 4.3.2.1.2, and 4.3.2.2.2 would be revised to clarify the term ''total bypass function.''

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. The amendment application proposes to add a provision to TS Table 3.3–3, Safety Features Actuation System (SFAS) Instrumentation, and TS Table 3.3–11, Steam and Feedwater

Rupture Control System (SFRCS) Instrumentation, to permit certain SFAS and SFRCS instrument channels to [be] placed in an inoperable condition for up to 8 hours during surveillance testing without declaring the channel inoperable and entering the Action statement. This proposed change would reduce burden placed on the control room operators and is essentially administrative in nature. The proposed change to the TS Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System instrumentation, is associated with the changes to TS Tables 3.3–3 and 3.3–11. These changes will not significantly change testing methodology, system unavailability, or system reliability. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report (USAR).

The proposed changes to Limiting Condition for Operation 3.3.1.1, Surveillance Requirement (SR) 4.3.1.1.2, SR 4.3.2.1.2, and SR 4.3.2.2.2 to clarify the nomenclature of the Reactor Protection System (RPS), SFAS, and SFRCS bypass functions being tested are administrative in nature. These changes will not effect any plant hardware or the performance of any test. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS USAR.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation, or radiological releases are not affected by the proposed changes. Existing system and component redundancy is not affected by the proposed changes. The existing system and component operation is not affected by the proposed changes, and the assumptions used in evaluating the radiological consequences in the DBNPS USAR are not invalidated. Therefore, for each postulated accident the consequences remain bounded by the consequences from the previously evaluated accidents.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these proposed changes do not involve any physical changes to systems or components, nor do they alter the manner in which the systems or components are operated. No new or different accident initiators or equipment failure modes are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because, for the proposed changes, there are no new or significant changes to the initial conditions contributing to accident severity or consequences. Accordingly, there are no significant reductions in a margin of safety.

On the basis of the above, the DBNPS has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 1, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement (SR) 4.0.5, Applicability, TS Bases 4.0.5, and TS Bases 3/4.4.2 and 3/4.4.3, Reactor Coolant System—Safety Valves, Regarding Inservice Testing Requirements.

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. The amendment application proposes to revise DBNPS Technical Specification (TS) Surveillance Requirement 4.0.5, Applicability, and its associated Bases and TS Bases 3/4.4.2 and 3/4.4.3, Reactor Coolant System—Safety Valves. The proposed changes would modify the Technical Specifications to conform to the requirements of Section 50.55a(f) of Title 10 of the Code of Federal Regulations regarding the inservice testing of pumps and valves for the third and successive 120-month intervals. The current DBNPS TS reference the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) which is consistent with Section 50.55a(f).

In addition, surveillance interval definitions for ''semi-quarterly,'' ''every 9 months,'' and ''biennially or every 2 years,'' as used in the ASME Code would be added to TS 4.0.5.b to ensure consistent interpretation of the terms. The proposed changes do not affect any plant hardware and do not affect the probability of any equipment malfunction or accident-initiating event.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no equipment, accident conditions, or assumptions are affected which could lead to a significant increase in radiological consequences.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new or different accident initiators are introduced by these proposed changes.

3. Not involve a significant reduction in a margin of safety because there are no changes to the initial conditions contributing to accident severity or consequences. Consequently, there are no significant reductions in a margin of safety.

On the basis of the above, the DBNPS has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

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

NRC Section Chief: Anthony J. Mendiola.

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: April 17, 2001.

Description of amendment request: The proposed amendments would implement minor changes and corrections to the Technical Specifications (TS) to correct administrative errors (e.g., typographical, amendment tracking number, etc.), or to incorporate changes that have been justified by previously approved license amendments and should have been made as part of those submittals, or to correct logic errors (e.g., TS operating mode breakpoints based on pressurizer pressure and not temperature). Also, the proposed amendments would revise the Units 1 and 2 TS to delete obsolete terminology and provide conforming changes to reflect the recently implemented change to 10 CFR 50.59.

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

(1) Operation of the Facility in Accordance With the Proposed Amendment Would Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

These proposed license amendments require no plant hardware or operational modifications. The proposed changes either correct various administrative errors (e.g., typographical errors, amendment tracking number errors), incorporate changes that have been justified by previously approved license amendments and should have been made as part of those submittals, correct logic errors, or are necessary to implement the 10 CFR 50.59 rule change.

Therefore, 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.

(2) Operation of the Facility in Accordance with the Proposed Amendment Would Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

No modifications to either plant hardware or operational procedures are required to support these proposed license amendments; hence, no new failure modes are created. The proposed changes either correct various administrative errors (e.g., typographical errors, amendment tracking number errors), incorporate changes that have been justified by previously approved license amendments and should have been made as part of those submittals, correct logic errors, or are necessary to implement the 10 CFR 50.59 rule change.

Therefore, 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.

(3) Operation of the Facility in Accordance With the Proposed Amendment Would Not Involve a Significant Reduction in a Margin of Safety

The majority of TS corrections proposed by these license amendments are administrative in nature in that they either correct typographical errors (e.g., ODCM verses OCDM), are justified by previous license amendments (e.g., surveillance requirements for T_{hot} wide versus narrow range instrumentation), or correct logic errors (e.g., ECCS subsystem TS headings based on operating mode, with Mode 3 breakpoints based on pressurizer pressure and not temperature). The overly restrictive emergency power requirements for non critical single train quality related radiation monitors are being removed, while critical radiation monitor emergency power requirements are unaffected by the change.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Patrick M. Madden (Acting).

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

Date of amendment request: April 18, 2001.

Description of amendment request: The proposed amendment would implement an improved heat flux correlation (designated ABB–NV) previously approved by the NRC for Westinghouse-Combustion Engineering, as documented in the topical reportCENPD–387–P–A, Rev 000. The proposed change updates Technical Specification (TS) 6.9.1.11, ''Core Operating Limits Report (COLR),'' to include the topical report in the list of analytical methods used. Additionally, the Bases for TS 2.1.1, ''Reactor Core,'' would be modified to reflect use of the improved heat flux correlation.

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

The proposed amendment would allow the implementation of ABB–NV critical heat flux correlation to St. Lucie Unit 2 core. The proposed changes have no adverse impact on the operation of the plant and have no relevance to the accident initiators. There are no changes to the plant configuration, and thus the frequency of occurrence of previously analyzed accidents is not affected by the proposed changes. With the application of the added methodology (the approved ABB–NV DNB correlation), the safety analysis would continue to remain consistent with the design basis requirements. The proposed changes, including changes to the TS Bases, have no adverse effect on the safety analysis and thus would not involve a significant increase in the consequences of design basis accidents. Changes to the COLR limits will continue to be controlled per Generic Letter 88–16 under the provisions of 10 CFR 50.59 and the requirements of TS 6.9.1.11.c.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Use of the Modified Specification Would Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed amendment updates the list of approved methodology in TS 6.9.1.11 and makes corresponding changes to the TS Bases for TS 2.1.1. These changes would not create the possibility of a new kind of accident since there is no change to plant configuration, systems, or components, which would create new failure modes. The modes of operation of the plant would remain unchanged.

Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Use of the Modified Specification Would Not Involve a Significant Reduction in a Margin of Safety

The proposed changes have no significant adverse impact on the safety analysis. As such, these changes would continue to provide margin to the acceptance criteria for Specified Acceptable Fuel Design Limits (SAFDL), 10 CFR 50.46(b) requirements, primary and secondary overpressurization, peak containment pressure, potential radioactive releases, and existing limiting conditions for operation. The future use of updated approved methodology will follow all design basis requirements to ensure that a safety margin to the acceptance criteria would continue to remain available at all power levels for operation of St. Lucie Unit 2.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Patrick M. Madden (Acting).

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

Date of amendment request: March 29, 2001.

Description of amendment request: The licensee proposed to amend the Technical Specifications (TSs) in three areas, adopting three NRC-approved Technical Specification Task Force (TSTF) issues. This notice is concerned with changes covered by one of the three issues, identified as TSTF–51.

The licensee proposed to adopt TSTF–51, reducing the operability requirements for certain engineered safeguard features (ESFs) such as secondary containment, standby gas treatment, control room envelop filtration. The current requirements specify that these ESFs be operable during movement of irradiated fuel in the secondary containment, and during core operation. The proposed changes would specify these ESFs be operable during movement of recently irradiated fuel in the secondary containment, and would eliminate the applicability during core alteration. The associated licensee-controlled TS Basis document would also be changed to reflect the changes.

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

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. TSTF–51 involves no hardware design change, thus there will be no adverse effect on the functional performance of the ESFs to mitigate accident consequences. The ESFs are not initiators of any previously analyzed accidents, thus the proposed changes cannot increase the probability of any previously analyzed accidents. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. TSTF–51 involves no hardware design change or procedural change; hence all components, systems, and structures will continue to perform as originally designed by the licensee and previously accepted by the NRC staff. Therefore, the proposed changes covered by TSTF–51 will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since TSTF–51 involves no change to the design, operational procedure, or analysis methodology, TSTF–51 will not affect in any way the performance characteristics and original intended functions of any system, structure or component. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the part of the amendment request identified as TSTF–51 involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: March 29, 2001.

Description of amendment request: The licensee proposed to amend the Technical Specifications (TSs) in three areas, adopting three NRC-approved Technical Specification Task Force (TSTF) issues. This notice is concerned with one of the three changes, identified as TSTF–204.

The licensee proposed to adopt TSTF–204, revising Limiting Condition for Operation (LCO) 3.8.5. Currently, LCO 3.8.5 requires that direct current (DC) power subsystems shall be OPERABLE to support the electrical power distribution subsystems required by LCO 3.8.9 (pertaining to shutdown conditions). Adoption of TSTF–204 would change this to require either the Division 1 or Division 2 DC electrical power subsystems, in addition to the Division 3 DC electrical power subsystem, shall be OPERABLE. This change would restore the TS to what it was before the TS was converted to the Improved TS format by Amendment No. 91 (February 15, 2000). The associated licensee-controlled TS Basis document would also be changed to reflect the changes.

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

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes to adopt TSTF–204 involve no hardware design change or operational procedure change, thus there will be no adverse effect on the functional performance of any plant SSC; the decreased operability requirement pertains to times when there is less demand on the electrical subsystems (i.e., during shutdown conditions). All structures, systems and components (SSCs) will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. Accordingly, the proposed operability requirements will lead to no significant increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. TSTF–204 involves no hardware design change or procedural change; hence all SSCs will continue to perform as originally designed by the licensee and previously accepted by the NRC staff. Therefore, the proposed changes covered by TSTF–204 will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since TSTF–204 involves no change to the design, operational procedure, or analysis methodology, TSTF–204 will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the part of the amendment request identified as TSTF–204 involves no significant hazards consideration.

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

NRC Section Chief: Richard Correia, Acting.

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

Date of amendment request: March 29, 2001.

Description of amendment request: The licensee proposed to amend the Technical Specifications (TSs) in three areas, adopting three NRC-approved Technical Specification Task Force (TSTF) issues. This notice is concerned with one of the three changes, identified as TSTF–287.

The licensee proposed to adopt TSTF–287, adding to Section 3.7.2, Control Room Envelope Filtration System (CREFS), a note to permit the control room envelope be opened intermittently under administrative control, and a new Condition B allowing 24 hours to restore operability of the two CREFS subsystems if their operability is lost due to inoperable control room envelope boundary. These proposed provisions would allow time to diagnose, plan and possibly repair, and test most problems with the control room envelope boundary. The associated licensee-controlled TS Basis document would also be changed to reflect the TS changes.

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

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change to adopt TSTF–287 involves no hardware design change or operational procedure change, thus there will be no adverse effect on the functional performance of any plant structures, systems or components (SSCs). The allowance to open the control room envelope intermittently does not increase accident consequences on control personnel since the administrative controls would rapidly restore integrity. Allowing 24 hours to restore the integrity of the control room envelope could result in an increase in consequences of a design-basis accident occurring during this time to control room personnel, but the administrative controls in place would easily and quickly reverse the condition, re-establishing control room envelope integrity, and thus limiting increases in consequences. Thus, all SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. Accordingly, the proposed operability requirements will lead to no significant increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. TSTF–287 involves no hardware design change or procedural change; hence it does not negatively affect the design or performance of any SSC, and all SSCs will continue to perform as originally designed by the licensee and previously accepted by the NRC staff. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since TSTF–287 involves no change to the design, operational procedure, or analysis methodology, TSTF–287 will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the part of the amendment request identified as TSTF–287 involves no significant hazards consideration.

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

NRC Section Chief: Richard Correia, Acting.

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

Date of amendment request: May 2, 2001.

Description of amendment request: The proposed amendment would (1) relocate requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section XI, inservice testing (IST) program currently contained in technical specification surveillance requirement (TSSR) 4.15.B to the TS Administrative Control Section 6.8, Programs and Manuals, (2) make conforming changes to several surveillance requirements to reflect the change in reference from TSSR 4.15.B to the licensee-controlled IST Program, (3) reword TSSRs 4.5.A.3 and 4.5.D.1 to be consistent with NUREG–1433, (4) incorporate TS Task Force (TSTF) initiative TSTF–279 into TS Administrative Control Section 6.8, and (5) revise TSSRs 4.6.H.1, 4.6.H.3, and Table 4.6.1 to change the inspection and functional testing interval extensions reference from plus-or-minus 25 percent to plus 25 percent.

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

1. The Proposed Amendment Will Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The requested changes are administrative in nature in that they relocate IST [inservice testing] requirements from the Monticello TS [Technical Specifications] to a licensee controlled IST program, rewrite TS Surveillance Requirements 4.5.A.3 and 4.5.D.1 for clarification using the wording from NUREG–1433 and revise TS surveillance requirements for inspection and functional testing interval extensions. The requested changes will not revise previous commitments to 10 CFR 50.55a of [sic] ASME [American Society of Mechanical Engineers] Code [ASME Boiler and Pressure Vessel Code], Section XI, IST requirements.

The proposed changes do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Since the accident analyses remain bounding, their radiological consequences are not adversely affected.

Therefore, the probability or consequences of an accident previously evaluated are not affected.

2. The Proposed Amendment Will Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Analyzed

The requested changes are administrative in nature in that they relocate IST requirements from the Monticello TS to the licensee controlled IST program, rewrite TS Surveillance Requirements 4.5.A.3 and 4.5.D.1 for clarification using the wording from NUREG–1433 and revise TS surveillance requirements for inspection and functional testing interval extensions. The requested changes will not revise previous commitments to 10 CFR 50.55a or ASME Code, Section XI, IST requirements.

The proposed changes do not involve changes to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes.

Therefore, the possibility of a new or different kind of accident from any accident previously analyzed is not created.

3. The Proposed Amendment Will Not Involve a Significant Reduction in the Margin of Safety

The requested changes are administrative in nature in that they relocate IST requirements from the Monticello TS to the licensee controlled IST program, rewrite TS Surveillance Requirements 4.5.A.3 and 4.5.D.1 for clarification using the wording from NUREG–1433 and revise TS surveillance requirements for inspection and functional testing interval extensions. The requested changes will not revise previous commitments to 10 CFR 50.55a or ASME Code, Section XI, IST requirements. Program requirements will remain to ensure that Code requirements are met.

Therefore, a significant reduction in the margin of safety is not involved.

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.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: April 11, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to relax the frequency for testing of excess flow check valves (EFCVs). Specifically, TS surveillance requirement 4.6.3.4 would be changed to revise required testing of EFCVs from once per 18 months for all valves to a test of a representative sample each 18 months such that all valves are tested once in 10 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. The NRC staff's review is presented below:

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

The proposed changes affect the surveillance interval for the EFCV's, allowing a reduced number of valves to be tested at each interval. There are no physical plant modifications associated with this change. The EFCV's, which are installed on instrument lines penetrating containment, are designed to close in order to isolate containment upon a failure of the instrument line downstream of the valve. Since the EFCV's are designed to provide an accident mitigation function (i.e., minimize radiological effects due to an instrument line break), their postulated failure to close as a result of the proposed reduced testing frequency is not considered an initiator to any previously evaluated accidents. Therefore, there is no increase in the probability of occurrence of an accident as a result of the proposed changes.

The design basis analyses for an instrument line break is evaluated in Section 15.6.2 of the Hope Creek Updated Final Safety Analysis Report (UFSAR). These analyses do not take credit for the closure of the EFCV's. The postulated failure of an EFCV to close as a result of the proposed reduced testing frequency is bounded by the existing UFSAR analyses. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes in TS surveillance requirements allow a reduced number of EFCV's to be tested each operating cycle. No other changes are being requested. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. These changes will not alter any process variables, structures, systems, or components as described in the safety analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The radiological consequences of an unisolable break of an instrument line have previously been evaluated in Section 15.6.2 of the Hope Creek UFSAR. The accident analyses assume that the line break results in the release of reactor coolant into the Reactor Building until the reactor pressure vessel is depressurized. The analyses do not take credit for the closure of the EFCV's. The proposed reduced testing frequency only changes the potential for an undetected failure of an EFCV and does not change the event sequence upon which the current safety margin related to radiological consequences is based. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: April 18, 2001.

Description of amendment request: South Carolina Electric & Gas Company (SCE&G) proposes a change to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirements to revise the volumetric flow units for TS 4.7.6.c.1, c.3, e.1, e.3, and f to identify standard flow units expressed as standard cubic feet per minute. Volumetric flow units for TS 4.6.3.b.1, b.2, c.1, d, and g, and TS 4.9.11.b.1, b.3, d.1, e, and f are being revised to identify actual air flow units and are expressed as actual cubic feet per minute.

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:

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated? Changes associated with the identification of proper flow units are editorial and have no impact.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated? Changes associated with the identification of proper flow units are editorial and have no impact.

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

safety for any of the ventilation systems associated with the proposed change is not compromised. Changes associated with the identification of proper flow units are editorial and have no impact.

There are no significant safety hazards created by the change. There is no new or different accident postulated since the change is considered editorial. The design requirements of Regulatory Guide 1.52 remain satisfied. Therefore, there is no significant decrease in the margin of safety.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: May 14, 2001 (TS 01–02).

Brief description of amendments: The proposed amendment would change the Sequoyah (SQN) Unit 1 Operating License Condition 2.C.(9)(d) to clarify the lower voltage threshold for eddy current inspections.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), Tennessee Valley Authority (TVA), 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 change does not alter plant equipment, system design, or operating practices. The clarification of SQN's Unit 1 [steam generator] SG inspection commitment provides a conservative inspection strategy that defines 1 volt as the lower threshold. The 1-volt threshold is based on the subjectivity uncertainties associated with interpreting bobbin coil probe data to distinguish a dent below 1 volt. Given the current capability of eddy current technology, TVA's proposed change will define a reasonable criteria for tube inspection.

TVA's proposed change continues to ensure that structural and leakage integrity of SQN's Unit 1 SG tubes is maintained. Accordingly, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the SQN Final Safety Analysis Report.

B. The Proposed Amendment Does Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

SQN limits SG tube leakage between the primary coolant system and the secondary coolant system to 150 gallons per day per SG. This leakage limit ensures that tube cracks have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. In addition, inservice inspections are performed in accordance with Regulatory Guide 1.83, Revision 1, ''Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes,'' to ensure that structural integrity of SG tubes is maintained during the plant operation cycle.

The proposed change does not modify plant equipment, system design, or operating practices. The clarification of SQN's Unit 1 SG inspection commitment provides an inspection strategy that defines a minimum ''calling'' threshold for dent inspection. The 1-volt threshold is an inspection strategy based on the subjectivity associated with interpreting bobbin coil probe data below 1 volt for dented intersections. TVA's proposed change will continue to provide conservative inspection criteria that maintains structural and leakage integrity of SQN's Unit 1 SG tubes.

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

C. The Proposed Amendment Does Not Involve a Significant Reduction in a Margin of Safety

TVA's proposed clarification of the 1-volt threshold will continue to provide a conservative inspection criteria that will ensure that SG tube structural and leakage integrity is maintained. Accordingly, the margin of safety is not reduced.

The NRC 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: Patrick M. Madden (Acting).

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

Date of amendment request: April 25, 2001.

Brief description of amendments: The proposed changes would revise Technical Specification (TS) 3.8.1, ''AC [alternating current] Sources—Operating,'' to extend the allowable Completion Times for the Required Actions associated with restoration of an inoperable Emergency Diesel Generator (EDG) and an inoperable offsite circuit (i.e., startup transformer). In addition, the TS Surveillance Requirement (SR) corresponding to the 24-hour EDG endurance run in SR 3.8.1.14 would be revised to allow the SR to be performed during Modes 1 and 2. The proposed changes would also revise TS 3.8.9, ''Distribution Systems—Operating,'' to extend the allowable Completion Times for the Required Actions associated with restoration of an inoperable AC electrical power distribution subsystem (i.e., 6.9 kilovolt (kV) AC safety bus).

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. Do the Proposed Changes Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

Response: No.

The proposed Technical Specification changes do not significantly increase the probability of occurrence of a previously evaluated accident because the 6.9 kV AC components (i.e., Emergency Diesel Generators (EDGs), startup transformers (STs), and safety-related (Class 1E) busses) are not initiators of previously evaluated accidents involving a loss of offsite power. The proposed changes to the Technical Specification Action Completion Times do not affect any of the assumptions used in the deterministic or the Probabilistic Safety Assessment (PSA) analysis.

The proposed Technical Specification changes will continue to ensure the 6.9 kV AC components perform their function when called upon. Extending the Technical Specification Completion Times to 14 days and allowing the performance of the EDG 24-hour run test in either MODES 1 or 2 does not affect the design of the EDGs, the operational characteristics of the EDGs, the interfaces between the EDGs and other plant systems, the function, or the reliability of the EDGs. Thus, the EDGs will be capable of performing either accident mitigation function and there is no impact to the radiological consequences of any accident analysis.

To fully evaluate the effect of the changes to the 6.9 kV AC components, Probabilistic Safety Analysis (PSA) methods and deterministic analysis were utilized. The results of this analysis show no significant increase in the Core Damage Frequency.

The Configuration Risk Management Program (CRMP) in Technical Specification 5.5.18 is an administrative program that assesses risk based on plant status. Adding the requirement to implement the CRMP for Technical Specification 3.8.1 and 3.8.9 requires the consideration of other measures to mitigate consequences of an accident occurring while a 6.9 kV AC component is inoperable.

The proposed changes do not alter the operation of any plant equipment assumed to function in response to an analyzed event or otherwise increase its failure probability. Therefore, these changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Do the Proposed Changes Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

Response: No.

These proposed changes do not change the design, configuration, or method of operation of the plant. The proposed activities involves [sic] a change to the allowed plant mode for the performance of specific Technical Specification surveillance requirements. No physical or operational change to the 6.9 kV AC components or supporting systems are made by this activity. Since the proposed changes do not involve a change to the plant design or operation, no new system interactions are created by this change. The proposed Technical Specification changes do not produce any parameters or conditions that could contribute to the initiation of accidents different from those already evaluated in the Final Safety Analysis Report.

The proposed changes only address the time allowed to restore the operability of the 6.9 kV AC components. Thus the proposed Technical Specification changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the Proposed Changes Involve a Significant Reduction in a Margin of Safety?

Response: No.

The proposed changes do not affect the Limiting Conditions for Operation or their Bases that are used in the deterministic analysis to establish any margin of safety. PSA evaluations were used to evaluate these changes, and these evaluations determined that the net changes are either risk neutral or risk beneficial. The proposed activities involves [sic] changes to certain Completion Times and to the allowed plant mode for the performance of specific Technical Specification Requirements. The proposed changes remain bounded by the existing Surveillance Requirement Completion Times and therefore have no impact to the margins of safety.

The proposed change does [sic] not involve a change to the plant design or operation and thus does not affect the design of the 6.9 kV AC components, the operation characteristics of the 6.9 kV AC components, the interfaces between the 6.9 kV AC components and other plant systems, or the function or reliability of the 6.9 kV AC components. Because 6.9 kV AC components performance and reliability will continue to be ensured by the proposed Technical Specification changes, the proposed changes do not result in a reduction in the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 23, 2001.

Description of amendment request: The amendment would update the facility operating license (FOL) by deleting obsolete information, correcting errors, and making administrative changes to enhance the context and provide consistency.

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 Operation of Vermont Yankee Nuclear Power Station in Accordance With the Proposed Amendment Will Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change makes editorial changes and brings the FOL up to date with the expectations of Massachusetts regulatory agencies. Since reactor operation under the proposed amendment is unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events or assumed mitigation of accident or transient events. The structural and functional integrity of plant systems is unaffected. Thus, there is no significant increase in the probability or consequences of accidents previously evaluated.

2. The Operation of Vermont Yankee Nuclear Power 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

The proposed change does not affect any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created. No safety-related equipment or safety functions are altered as a result of these changes. Because it does not involve any change to the plant or the manner in which it is operated, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The Operation of Vermont Yankee Nuclear Power Station in Accordance With the Proposed Amendment Will Not Involve a Significant Reduction in a Margin of Safety

The proposed change does not affect design margins or assumptions used in accident analyses, and has no effect on any assumed analysis initial condition. The capability of safety systems to function and limiting safety system settings are similarly unaffected as a result of this change. Thus, the proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Public Reading Room).

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

Date of application for amendment: December 20, 2000, as supplemented March 14, 2001.

The March 14, 2001, letter provided additional clarifying information which did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice. A March 23, 2001, letter provided a camera-ready copy of the revised technical specification pages.

Brief description of amendment: The amendment allows the expanded use of the Framatome Cogema Fuels M5 alloy for fuel rod cladding and fuel assembly spacer grids. A related Bases change is included with the licensee's application.

Date of issuance: May 10, 2001.

Effective date: As of the date of issuance and shall be implemented no later than the startup of Cycle 14 operation, approximately October 1, 2001.

Amendment No.: 233.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.*Date of initial notice in* ***Federal Register:*** February 6, 2001 (66 FR 9379).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 13, 2000.

Brief description of amendments: The amendment revised the technical specifications to delete the ''Power Range Neutron Flux High Negative Rate,'' Trip Function from Reactor Trip System Instrumentation. The changes allow elimination of this unnecessary function and thereby reduces the potential for a transient. The changes are consistent with the Westinghouse Topical report previously accepted by the NRC.

Date of issuance: May 17, 2001.

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

Amendment Nos.: 114, 114, 120, 120.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11054).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 17, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 21, 2000, as supplemented April 25, April 26, May 3 (two letters), and May 8, 2001.

Brief description of amendment: Amendment conforms the license to reflect the transfer of operating authority under Operating License No. DPR–20 to Nuclear Management Company, LLC, as approved by order of the Commission dated April 19, 2001 (66 FR 21021 dated April 26, 2001).

Date of issuance: May 15, 2001.

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

Amendment No.: 201.

Facility Operating License No. DPR–20. Amendment revised the Operating License.

Date of initial notice in ***Federal Register:*** December 19, 2000 (65 FR 79431).

The supplemental letters dated April 25, April 26, May 3 (two letters), and May 8, 2001, were within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 7, 2000.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) in accordance with changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1, made by the Nuclear Energy Institute Technical Specifications Task Force Change Number 258, Revision 4, addressing changes to various administrative controls in the TSs.

Date of issuance: May 3, 2001.

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

Amendment No.: 196.

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

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7678).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 7, 2000.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) in accordance with changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1, made by the Nuclear Energy Institute Technical Specifications Task Force Change Number 287, Revision 5, addressing allowances for breach of the control room envelope. Also, the action table for TS Limiting Condition for Operation 3.7.10 is corrected by restoring Required Action D.2 (now renumbered to E.2), which was inadvertently omitted in Amendment No. 189, issued on November 30, 1999.

Date of issuance: May 3, 2001.

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

Amendment No.: 197.

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

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7678).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 7, 2000.

Brief description of amendment: The amendment changes Technical Specification (TS) 3.5.2 in accordance with changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1, made by the Nuclear Energy Institute Technical Specifications Task Force change number 325, Revision 0, addressing changes to the structure of the TS Limiting Condition for Operation (LCO) for the Emergency Core Cooling System.

Date of issuance: May 3, 2001.

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

Amendment No.: 198.

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

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7675).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 6, 2001, as supplemented by letter dated May 3, 2001.

Brief description of amendment: The amendment revises the surveillance requirements pertaining to testing of the emergency diesel generators. The change removes the restrictions in plant technical specifications that prohibit performing the required testing during plant operation (Modes 1, 2, and 3). Additionally, the amendment modifies plant technical specifications to allow the endurance test to be performed in lieu of the load-run test provided the requirements of the load-run test, except the upper limit, are met.

Date of issuance: May 18, 2001.

Effective date: May 18, 2001, to be implemented within 30 days of the date of issuance.

Amendment No.: 173.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 17, 2001 (66 FR 19801).

The May 3, 2001, supplemental letter provided clarifying information, did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 18, 2001.

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: February 5, 2001, as supplemented on April 13, 2001.

Brief description of amendment: This amendment changes the Safety Limit Minimum Critical Power Ratio in Technical Specification (TS) 2.1.2 from 1.08 to 1.06. The amendment makes administrative changes to TS 5.6.5, ''Core Operating Limits Report,'' section a and b. The amendment makes administrative changes to Bases section 2.1 to reflect this TS change and to Bases section 3.11 to reflect an earlier TS change.

Date of issuance: May 8, 2001.

Effective date: As of the date of issuance, and shall be implemented prior to startup from Refueling Outage 13.

Amendment No.: 191.

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

Date of initial notice in ***Federal Register:*** March 7, 2001 (66 FR 13802).

The April 13, 2001, letter provided clarfying information that was within the scope of the amendment request and did not change the proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 9, 2000.

Brief description of amendment: This amendment revised the technical specifications by approving thirteen of the simpler, generic administrative/editorial/consistency improvements agreed upon between the Nuclear Energy Institute (NEI) Technical Specification Task Force (TSTF) and the Nuclear Regulatory Commission (NRC), subsequent to the conversion of the PNPP Technical Specifications to the improved Standard Technical Specifications. The improvements include TSTF–5, TSTF–32, TSTF–38, TSTF–52, TSTF–65, TSTF–104, TSTF–106, TSTF–118, TSTF–152, TSTF–166, TSTF–258, TSTF–278, and TSTF–279.

Date of issuance: May 15, 2001

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

Amendment No.: 120

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

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77920).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 12, 2001.

Brief description of amendments: The amendments reduced the requirement for average reactor coolant temperature during the rod cluster control assembly drop test from greater than or equal to 541°F to greater than or equal to 500°F.

Date of issuance: May 7, 2001.

Effective date: May 7, 2001.

Amendment Nos: 214 and 208.

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

Date of initial notice in ***Federal Register:*** April 4, 2001 (66 FR 17967).

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

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: June 9, 2000.

Brief description of amendment: The amendment revised the Technical Specifications (TS) 3.7.7 for the Main Turbine Bypass Valve surveillance test frequency, TS surveillance requirement SR 3.7.7.1 frequency from 31 days to 92 days.

Date of issuance: May 16, 2001.

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

Amendment No.: 239.

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

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46009).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, 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 20, 2000, as supplemented February 6 and May 3, 2001.

Brief description of amendments: These amendments incorporate changes to the Technical Specifications to increase the allowable deviation in individual rod position indication. By the February 6, 2001, supplemental letter, the licensee withdrew portions of the original application that dealt with operation at greater than 85-percent power. The licensee plans to submit those portions that deal with operation at greater than 85-percent power as a separate amendment request at a later time.

Date of issuance: May 8, 2001.

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

Amendment Nos.: 200 and 205.

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

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9386).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 8, 2000, as supplemented by letter dated January 4, 2001.

Brief description of amendments: The amendments revised Technical Specification (TS) Section 3.5.5, ''Emergency Core Cooling Systems—Seal Injection Flow,'' to replace the description of the seal injection flow with a description consistent with the method used to establish and verify reactor coolant pump seal injection flow limits and the method used to calculate the seal injection flow in the safety analyses for the Diablo Canyon Nuclear Power Plant.

Date of issuance: May 7, 2001.

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

Amendment Nos.: Unit 1—148; Unit 2—148.

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

Date of initial notice in ***Federal Register:*** April 4, 2001 (66 FR 17968).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 28, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) Section 5.5.2.b, ''Primary Coolant Sources Outside Containment'' by changing the system leak test frequency from ''at refueling cycle intervals or less'' to ''at least once every 18 months.'' The proposed change will also allow the provisions of Surveillance Requirement (SR) 3.0.2 to apply to TS Section 5.5.2.b.

Date of issuance: May 11, 2001.

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

Amendment Nos.: 119 and 97.

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

Date of initial notice in ***Federal Register:*** September 20, 2000 (65 FR 56955).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 11, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 5, 2000.

Brief description of amendments: The amendments revise the licenses to reflect changes to the Updated Final Safety Analysis Report due to revisions to the dose equivalent iodine analysis.

Date of issuance: May 11, 2001.

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

Amendment Nos.: 120 and 98.

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

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 11, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 6, 2000, as supplemented by letter dated February 9, 2001.

Brief description of amendments: The amendments revised Technical Specifications (TS) 3.7.10, ''Control Room Emergency Filtration System (CREFS)—Both Units Operating,'' TS 3.7.11, ''Control Room Emergency Filtration System (CREFS)—One Unit Operating,'' and TS 3.7.13, ''Piping Penetration Area Filtration and Exhaust System (PPAFES),'' to establish actions to be taken for inoperable ventilation systems due to a degraded control room pressure boundary or piping penetration area pressure boundary, respectively. Specifically, the changes allow the pressure boundaries of ventilation systems such as CREFS and PPAEFS to be opened intermittently under administrative control. A new condition is also added that allows 24 hours to restore inoperable CREFS and PPAFES pressure boundaries before requiring the units to perform an orderly shutdown. The applicable TS Bases have been revised to document these TS changes and to provide supporting information. These changes are based on Technical Specifications Task Force (TSTF) –287, Revision 5, to the Standard Technical Specifications.

Date of issuance: May 14, 2001.

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

Amendment Nos.: 121 and 99.

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

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77926).

The supplemental letter dated February 9, 2001, provided clarifying information that did not change the scope of the November 6, 2000, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 14, 2001.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendment: March 9, 2001.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) by allowing insertion of up to four lead test assemblies containing downblended uranium, in accordance with Framatome Cogema Fuels Topical Report BAW 2328, into the Sequoyah Unit 1 core for up to two fuel cycles.

Date of issuance: May 9, 2001.

Effective date: May 9, 2001.

Amendment No.: 268.

Facility Operating License No. DPR–77: Amendment revised the TSs.

Date of initial notice in ***Federal Register:*** April 4, 2001 (66 FR 17970).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 7, 2000.

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

Date of issuance: May 14, 2001.

Effective date: May 14, 2001.

Amendment Nos.: 225 and 225.

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

Date of initial notice in ***Federal Register:*** March 21, 2001, (66 FR 15931), supersedes March 20, 2000 (65 FR 15388).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 14, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 22, 2000.

Brief description of amendment: The amendment relocated certain administrative requirements from the Yankee Nuclear Power Station (YNPS) Defueled Technical Specifications to the YNPS Decommissioning Quality Assurance Program. Additional editorial changes to titles and designations were also made.

Date of issuance: May 15, 2001.

Effective date: May 15, 2001.

Amendment No.: 155.

Facility Operating License No. DPR–3. The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 4, 2001 (66 FR 17972).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 22nd day of May 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–13400 Filed 5–29–01; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[File No. 1–12514]

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration; (Keystone Property Trust, Common Stock, Par Value $.01 Per Share)

May 23, 2001.

Keystone Property Trust, a Maryland real estate investment trust (''Issuer''), has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 12d2–2(d) hereunder,[2] to withdraw its Common Stock, par value $.01 per share (''Security''), from listing and registration on the American Stock Exchange (''Amex'').

The Issuer stated in its application that it has met the requirements of Amex Rule 18 by complying with all applicable laws in effect in the State of Maryland, in which it is incorporated, and with the Amex's rules governing an issuer's voluntary withdrawal of a security from listing and registration. The Amex has in turn informed the Issuer that its does not object to the proposed withdrawal of the Issuer's Security from listing and registration on the Exchange.

The Board of Trustees (''Board'') approved a resolution on April 17, 2001 to withdraw the Issuer's Security from listing on the Amex and to list such Security on the New York Stock Exchange, effective May 9, 2001. The Issuer stated that the Board took such action in order to increase the profile and visibility of the Issuer in the public markets and to attract more interest in the Issuer from individuals and institutional investors.

The Issuer's application relates solely to the withdrawal of the Security from listing and registration on the Amex and shall have no effect upon the Security's continued listing and registration on the NYSE under section 12(b) of the Act.[3]

Any interested person may, on or before June 13, 2001, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549–0609, facts bearing upon whether the application has been made in accordance with the rules of the Amex and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

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

Jonathan G. Katz,

Secretary.

[FR Doc. 01–13527 Filed 5–29–01; 8:45 am]

BILLING CODE 8010–01–M

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

[2] 17 CFR 240.12d2–2(d).

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

[4] 17 CFR 200.30–3(a)(1).

SECURITIES AND EXCHANGE COMMISSION

[File No. 1–15237]

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration; (OTR Express, Inc., Common Stock, $.01 Par Value)

May 23, 2001.

OTR Express, Inc., a Kansas corporation (''Issuer''), has filed an