implementing the revised 10 CFR 50.59 requirements is June 1, 2001.

Need for Proposed Action: The applicant wants the implementation date of 10 CFR 50.59 and 10 CFR 72.48 to coincide. The applicant stated in the February 9, 2001, submittal that administering separate programs for less than a two month period to satisfy the current 10 CFR 72.48 schedule could become burdensome and create confusion.

Environmental Impacts of the Proposed Action: There are no significant environmental impacts associated with the proposed action. The new revision of 10 CFR 72.48 is considered less restrictive than the current requirements, with the exception of the additional reporting requirements. Continued implementation of the existing 10 CFR 72.48 until June 1, 2001, is acceptable to the NRC as stated in Regulatory Issues Summary 2001–03 which states that it is the NRC's view that both the old rule and the new rule provide an acceptable level of safety. Extending the current requirements until June 1, 2001, has no significant impact on the environment.

Alternative to the Proposed Action: Since there are no environmental impacts associated with the proposed action, alternatives are not evaluated other than the no action alternative. The alternative to the proposed action would be to deny approval of the scheduler exemption and, therefore, not allow SNC to implement the revised 10 CFR 72.48 requirements on the desired date, June 1, 2001. However, the environmental impacts of the proposed action and the alternative would be the same.

Agencies and Persons Consulted: On March 9, 2001, Georgia state official, Mr. James Hardeman, Environmental Radiation Program Manager, Georgia Department of Natural Resources, Environmental Protection Division, was contacted regarding the environmental assessment for the proposed action and had no comment.

Finding of No Significant Impact

The environmental impacts of the proposed action have been reviewed in accordance with the requirements set forth in 10 CFR part 51. Based upon the foregoing EA, the Commission finds that the proposed action of granting an exemption from 10 CFR 72.48, so that SNC may implement the amended requirements on June 1, 2001, will not significantly impact the quality of human environment. Accordingly, the Commission has determined that an environmental impact statement for the proposed action is not necessary.

The request for exemption was docketed under 10 CFR part 72, Docket 72–36. For further details with respect to this action, see the exemption request dated February 9, 2001, which is available for public inspection at the Commission's Public Document Room, One White Flint North Building, 11555 Rockville Pike, Rockville, Maryland 20852, or from the publicly available records component of NRC's agencywide documents access and management system (ADAMS).

ADAMS is accessible from the NRC web site at *http://www.nrc.gov/NRC/ADAMS/index.html* (the Public Electronic Reading Room).

Dated at Rockville, Maryland, this 5th day of April 2001.

For the Nuclear Regulatory Commission.

E. William Brach,

Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 01–9621 Filed 4–17–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 26 through April 6, 2001. The last biweekly notice was published on April 4, 2001 (66 FR 17962).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of 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 May 18, 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).

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: March 1, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) on page 4.5–3 to change the frequency of closure time testing of the main steam isolation valves (MSIVs). If approved, these tests would no longer occur during power operation. They would be conducted during each cold shutdown unless this test has been performed within the last 92 days.

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:

The proposed amendment does not:

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

The proposed change revises Technical Specification 4.5.F.3 to require MSIV full-stroke testing during each cold shutdown rather than quarterly at power. Since this change only affects the frequency of testing the isolation time of MSIVs, it does not impact the occurrence of accidents that the MSIVs are designed to mitigate. The 10% closure test that will be performed quarterly in order to test the MSIV closure scram instrumentation has some potential of causing an inadvertent closure of the MSIV. The current test of MSIV closure scram instrumentation is conducted with the quarterly full closure test and is performed at reduced power. The closure of an MSIV at the reduced power level does not result in a plant trip. Inadvertent closure of MSIVs is a transient of moderate frequency evaluated in the updated FSAR [final safety analysis report]. The small increase in potential for an MSIV full closure transient during the part-stroke test is offset by the decrease in potential transients due to the plant power manipulation necessary to perform the full closure test.

The proposed change affects the frequency of testing the MSIVs to ensure an acceptable level of reliability. Aligning the Oyster Creek test frequency for MSIVs with the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] and industry practice assures adequate reliability for valve closure. Therefore, the MSIVs will be capable of closing to mitigate accidents.

As a result of the discussion above, the change to the frequency of MSIV full closure testing does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There is no physical change in plant configuration associated with performing the MSIV full or partial closure tests. The MSIV closure scram is designed to anticipate the transient caused by valve closure with the plant in operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change affects the method of assuring the reliability of the MSIVs. The change from a quarterly full-stroke closure test at power to full-stroke tests during cold shutdowns combined with quarterly part-stroke tests to ensure instrument function provides adequate means of assuring MSIV operability. The reliability of MSIVs to close within the required 3–10 seconds has been consistently demonstrated and it is expected that the valves will continue to pass this test when done on a cold shutdown basis. The quarterly 10% closure reactor protection system testing will assure that the valves will respond to a closure signal.

Presently, the MSIVs are full-stroke closed quarterly at power in accordance with Technical Specification 4.5.F.3. The basis for the current quarterly full closure test at power and the proposed full closure test during cold shutdowns with part-stroking quarterly during instrument surveillance is consistent with the ASME Boiler and Pressure Vessel Code, Section XI. In addition, the proposed change is consistent with industry standard requirements contained in the Standard Technical Specifications, NUREG–1433, Revision 1. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Marsha Gamberoni.

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1 (Fermi1), Monroe County, Michigan.

Date of amendment request: November 6, 2000, (Reference NRC–00–0084) and supplement by letter dated March 12, 2001, (Reference NRC–01–0026).

Description of amendment request: The proposed amendment will revise the Technical Specifications by: (1) deleting Specification A.1, the definition of Physical Barrier; (2) deleting Specification A.2, the definition of Protected Area; (3) deleting Specification A.4, the definition of Authorized Person; (4) Specification A.7.a, change the words ''Protected Area'' to ''facility''; (5) Specification B.1, delete the discussion on method for controlling facility access and add words noting that the method for controlling access to the facility will be included in the Fermi 1 Safety Analysis Report; (6) deleting Specification C.1, Reactor Building Access which specifies access limitations to this building; (7) Specification C.2, change the words ''Protected Area'' to ''facility''; (8) deleting Specification E.1, Fuel and Repair Building which specifies access limitation to this building; (9) Specification H.4.a, change the words ''Protected Area'' to ''facility''; (10) Specification H.4.b, change the words ''Protected Area'' to ''facility''; (11) Specification I.6, deleting specific dosimetry requirements and replacing with a requirement that dosimetry will meet the requirements of 10 CFR Part 20; (12) Specification I.9.i, change the words ''Protected Area'' to ''facility''; and (13) Revise Figure B–1 to remove reference to the Protected Area Boundary and indication of structures that may be physically removed during the decommissioning process. In addition to these specific changes, the licensee will repaginate the Technical Specification. The above-listed are to support moving the licensee's program for controlling access to the Fermi 1 facility from the Fermi 1 license to the Fermi 1 Safety Analysis Report. This action would provide flexibility for the performance of decommissioning activities while maintaining controls commensurate with the small quantity of licensed material at Fermi 1 and its status.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

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

The changes to requirements regarding access to the facility and security do not affect the operation of any system. The requirements for having access control in the Fermi 1 Safety Analysis Report (F1SAR) will ensure people are aware when they are entering an area at Fermi 1 to which it is determined access should be controlled. The two analyzed accidents involved release of the activity in the liquid waste system and release of the activity in the residual sodium. These changes will not significantly increase the probability of these accidents since they do not affect operations of these systems. The possibility of an intruder entering the facility may slightly increase depending on the type of access controls implemented in the future which will be specified in the F1SAR. For example, if gates and doors are not closed and locked, but signs posted requiring permission for entry, the possibility of an unauthorized person entering the facility could increase. But, if such an intruder is intent on entering, he or she could do so under current access controls using common tools or equipment. The locks and barriers currently act as a reminder that the area is controlled. Future access control provisions will still provide that reminder.

Allowing the Protected Area to be different from the Restricted Area required by 10 CFR 20 will not increase the probability of an accident. The requirement for limiting access to a restricted area or areas will still be required by 10 CFR 20. This requirement is for personnel protection and is unrelated to accident probability.

Allowing the facility and the Protected Area required by 10 CFR Part 20 will not increase the probability of an accident. The requirement for limiting access to a restricted area or areas will still be required by 10 CFR Part 20. The requirement is for personnel protection and is unrelated to accident probability.

The change deleting the specific requirements for dosimetry will not affect the probability of an accident since they apply to monitoring of personnel not control or operation of the facility. The requirements for monitoring are in 10 CFR Part 20 and will be referenced in the Technical Specifications.

The radiological consequences of an accident will not be increased by the requested changes because the changes do not add radioactive material to the facility and the accidents analyses already assume release of all the activity in the primary sodium and liquid waste systems.

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

The requested changes do not change the method of operation of any system and so cannot create the possibility of a new or different accident. The analyzed accidents include release of all the radioactive material in the liquid waste system and release of all the radioactive material in the residual sodium remaining in the Protected Area which currently is the same as the licensed facility. Changing access, security, and boundary requirements cannot create a different type of accident.

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

The proposed changes could slightly reduce the margin of safety, but only from the perspective of making it easier for an unauthorized individual to enter the facility or Protected Area. A determined unauthorized person could violate existing requirements or proposed requirements to gain entry. However, the Technical Specifications will still require access control requirements for portions of the facility. Access control requirements described in the F1SAR will ensure that personnel know the Fermi 1 Protected Area is a controlled access area and will prevent anyone from unknowingly entering the portions of the facility for which access control is determined appropriate per the F1SAR. There is a limited amount of radioactive material remaining at Fermi 1. The requirement for a restricted area or areas, as appropriate, will still apply per 10 CFR Part 20 and is for individual protection not security. The proposed security measures are commensurate with the amount of radioactive material present in that neither 10 CFR Part 30 or 10 CFR Part 70 established a security requirement for the amount of radioactive material at Fermi 1.

Removing some buildings from Figure B–1 will not reduce the margin of safety. The figure will still fulfill the purpose of showing the facility boundary. The figure will still serve that purpose even if a building, structure, or barrier is modified or removed, since the provisions allowing changes requires that the boundary continues to encompass the area to which access is controlled. The accident analyses do not credit any building as a containment or as retaining any radioactive material during a possible accident. For the above reasons, the requested changes do not involve a significant reduction in margin of safety of Fermi 1.

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

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

NRC Branch Chief: Larry W. Camper.

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

Date of amendment request: January 24, 2001.

Description of amendment request: The amendment request proposes changes to the Technical Specifications (TSs) concerning certain operational conditions required when conducting core alterations or handling irradiated fuel in the primary containment. In addition, the licensee proposes to delete license condition 2.C.(17) and make certain editorial corrections.

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 relaxation of TS OPERABILITY requirements for containment and control room ventilation systems during specific shutdown conditions do not affect the probability of any accident previously evaluated and do not alter current accident analyses consequences. During plant shutdown, these systems and structures are accident mitigating features for the postulated Fuel Handling Accident (FHA) and are not considered the initiator to any previously analyzed accident. They need not be required during CORE ALTERATIONS because the only accident postulated to result in significant fuel damage and radiation release during shutdown conditions is the FHA. The control room filtration, inlet radiation detection, and the air conditioning systems will continue to be required during the handling of any irradiated fuel assembly and during operations with the potential for draining the reactor vessel (OPDRVs). The containment will only be required during OPDRVs and when moving recently irradiated fuel assemblies. The current FHA analysis of record (approved by Amendment 110) assumes the containment is open after the irradiated fuel has undergone a sufficient decay period (i.e., has not been part of a critical reactor core within the previous 11 days). The analysis demonstrates that the offsite doses remain well within the Standard Review Plan Guidelines (less than 25% of the 10 CFR [Part] 100 limits) and the control room doses remain less than the criteria of 10 CFR [Part] 50, Appendix A, General Design Criterion 19.

The proposed changes regarding the removal of the SGT [Standby Gas Treatment] system from the ''Primary Coolant Sources Outside Containment'' leakage control program does not affect the reliability or filtration efficiency of the SGT system. Current TS surveillances test filtration efficiency and secondary containment in-leakage. There are no unfiltered pathways to the suction of the fans that require leakage testing.

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

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

The proposed changes do not involve any design changes or any new modes of system operation. The proposed TS changes allow certain functions to be inoperable during CORE ALTERATIONS and during the handling of irradiated fuel that has undergone a sufficient radiation decay period. However, these out-of-service configurations are consistent with current design basis analyses. The removal of the SGT system from the ''Primary Coolant Sources Outside Containment'' leakage control program does not affect reliability or efficiency of the filtration system or otherwise affect the ability of the system to perform its safety function.

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

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

The proposed changes do not reduce the margin of safety, as defined by SRP [Standard Review Plan] 15.7.4 Rev 1. The only accident postulated to occur during shutdown that results in significant fuel damage and subsequent radiation release is the FHA. The offsite and control room doses due to a FHA with an open containment have previously been evaluated with conservative assumptions and that analysis is not affected by the proposed changes. The analysis demonstrates that due to radioactive decay following reactor shutdown, the primary containment function is only required when handling recently irradiated fuel.

The removal of the SGT system from the ''Primary Coolant Sources Outside Containment'' leakage control program does not affect reliability or efficiency of the filtration system or otherwise affect the ability of the system to perform its safety function.

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

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

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

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2), Shippingport, Pennsylvania

Date of amendment request: December 27, 2000.

Description of amendment request: The amendment request proposes various changes to the BVPS–1 and 2 technical specifications (TSs) and removal of a BVPS–1 license condition. The proposed BVPS–1 and 2 TS changes include (1) the revision of reactor trip system and engineered safety features actuation system instrumentation trip setpoints and allowable values; (2) the utilization of the Revised Thermal Design Procedure to generate additional departure from nucleate boiling (DNB) margin which facilitates revisions to the core safety limits, DNB parameters and Overtemperature and Overpower ΔT trip setpoints; (3) the relocation of certain requirements from the TS to the core operating limit report; (4) the relocation of certain requirements from the TS to the Licensing Requirements Manual; and (5) miscellaneous changes that improve internal consistency of the BVPS TSs, simplify the presentation of requirements, provide clarifications, and improve consistency with the improved standard TSs. Changes to the TS Bases in support of the TS changes are also proposed. In addition, the deletion of BVPs-1 license condition regarding limitations on less than 3-loop operation is included in the amendment request.

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

A. Revision to Setpoint and Allowable Values

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

The RPS [reactor trip system] and ESFAS [engineered safety feature actuation system] trip functions are part of the accident mitigation response and are not themselves an initiator for any transient. Therefore, the probability of an accident previously evaluated is not significantly affected.

This proposed amendment includes changes to RTS and ESFAS trip setpoints and allowable values that have been determined with the use of an approved methodology. The new values ensure that all automatic protective actions will be initiated at or before the condition assumed in the safety analysis. This change, which includes modification of the applicable Bases section(s), will allow the nominal trip setpoints to be adjusted within the calibration tolerance band allowed by the setpoint methodology. Plant operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of plant systems is unaffected. There will be no adverse effect on the ability of the channels to perform their safety functions as assumed in the safety analyses. Since there will be no adverse effect on the trip setpoints or the instrumentation associated with the trip setpoints, there will be no significant increase in the consequences of any accident previously evaluated.

The proposed amendment does involve a hardware change. The hardware change involves the deletion of f(ΔI) (BVPS Unit No. 1) and f2 (ΔI) (BVPS Unit No. 2) for the Overpower ΔT Trip Setpoint. This function is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint.

Other changes in trip system function, content and format are proposed based on the current configuration of the trip system hardware at BVPS Unit No. 1. Similarly, since the ability of the instrumentation to perform its safety function is not adversely affected, there will be no significant increase in the consequences of any accident previously evaluated.

The proposed editorial, administrative and format changes do not affect plant safety.

Therefore, this change does not involve any 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 amendment includes changes to the format and magnitudes of nominal trip setpoints and allowable values that preserve all safety analysis assumptions related to accident mitigation. The protection system will continue to initiate the protective actions as assumed in the safety analysis. The proposed changes to Limiting Safety System Settings (LSSS) 2.2.1 and LCO 3.3.2.1 will continue to ensure that the trip setpoints are maintained consistent with the setpoint methodology and the plant safety analysis. The proposed amendment does involve a hardware change. The hardware change involves the deletion of f(ΔI) (BVPS Unit No. 1) and f2 (ΔI) (BVPS Unit No. 2) for the Overpower ΔT Trip Setpoint. This function is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint. Therefore, this hardware change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Plant operation will not be changed.

Other proposed changes are made so that the technical specifications more accurately reflect the plant-specific trip system hardware in BVPS Unit No. 1.

Furthermore, the proposed changes do not alter the functioning of the RTS and ESFAS.

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

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

The proposed RTS and ESFAS trip setpoints are calculated with an approved methodology. The proposed changes to LSSS 2.2.1 and LCO 3.3.2.1 will continue to ensure that the trip setpoints are maintained consistent with the setpoint methodology and the plant safety analysis. Therefore, the response of the RTS and ESFAS to accident transients reported in the Updated Final Safety Analysis Report (UFSAR) is unaffected by this change. This proposed amendment does involve a hardware change. The hardware change involves the deletion of f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2) for the Overpower ΔT Trip Setpoint. This function is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint. Therefore, accident analysis acceptance criteria are not affected. Other proposed changes are made so that the protection system technical specifications more accurately reflect the plant-specific trip system hardware in BVPS Unit No. 1.

The proposed editorial, administrative, and format changes do not affect plant safety.

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

B. Revised Thermal Design Procedure (RTDP)—Overtemperature ΔT and Overpower ΔT

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

The revised Figure 2.1–1 and the Overtemperature and Overpower ΔT reactor trip functions do not involve a significant increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of all plant systems is unaffected. The Overtemperature and Overpower ΔT reactor trip functions are part of the accident mitigation response and are not themselves an initiator for any transient. Therefore, the probability of occurrence previously evaluated is not significantly affected.

The changes to Figure 2.1–1 and to the Overtemperature and Overpower ΔT reactor trip functions do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. Figure 2.1–1 provides restrictions to prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. It does not provide an automatic protective function but does provide the basis for the Overtemperature and Overpower ΔT reactor trip functions. These trip functions ensure that automatic protective actions will be initiated at or before the condition assumed in the safety analyses. These changes produce no adverse effect on the ability of these functions to perform their safety functions assumed in the safety analyses. In addition, the off-site mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the off-site dose predictions remain within the acceptance criteria of 10 CFR 100 limits for each of the transients affected. Since it has been concluded that the transient analyses results are unaffected by the parameter modifications, it is concluded that the probability or consequences of an accident previously evaluated are not significantly increased.

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 revised Figure 2.1–1 and Overtemperature and Overpower ΔT reactor trip functions do not create the possibility of a new or different kind of accident from any accident previously evaluated because the setpoint adjustments do not affect accident initiation sequences. No new operating configuration is being imposed by the setpoint adjustments that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

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

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

The changes to Figure 2.1–1 and to the Overtemperature and Overpower ΔT reactor trip functions do not involve a significant reduction in a margin of safety because the margin of safety associated with the Overtemperature and Overpower ΔT reactor trip functions, as verified by the results of the accident analyses, are within acceptable limits. All transients impacted by implementation of the RTDP methodology have been analyzed and have met the applicable accident analyses acceptance criteria. The margin of safety required for each affected safety analysis is maintained. This conclusion is not changed by the Overtemperature and Overpower ΔT setpoint modifications. The adequacy of the revised technical specification values to maintain the plant in a safe operating condition has been confirmed.

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

C. RTDP—Departure From Nucleate Boiling (DNB) Parameter Surveillance Requirements

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

The intent of the change is to preserve the Safety Analyses Limits for DNB (TS 3/4.2.5). There is no increase in the probability or consequences of an accident previously evaluated because there is no change to any design or analysis acceptance criteria. The structural and functional integrity of any plant system is unaffected. The proposed license amendment revises the surveillance requirement acceptance criteria for the DNB parameters. The indicated DNB parameters preserve the assumptions used in the accident analysis and, therefore, there is no significant increase in probability or consequences previously evaluated.

The changes to the DNB parameters do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. In addition, the off-site mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the off-site dose predictions remain within the limits of the 10 CFR 100 for each of the transients affected. Since it has been determined that the transient results are unaffected by these parameter modifications, it is concluded that the consequences of an accident previously evaluated are not significantly increased.

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 revised DNB parameter values do not create the possibility of a new or different kind of accident from any accident previously evaluated. The setpoint values do not affect the assumed accident initiation sequences. No new operating configuration is being imposed by changing these parameters that would create a new failure scenario. In addition, no new failure modes or single failures have been identified for any plant equipment. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

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

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

The DNB parameters are consistent with the UFSAR assumptions and maintain the required minimum DNBR [departure from nucleate boiling ratio] above the design limits throughout each analyzed transient. Thereby, the adequacy of the revised DNB parameter values to maintain the plant in a safe operating condition is confirmed.

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

D. Relocation to Colr [Core Operating Limits Report]

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

The proposed amendment is a programmatic and administrative change that does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The cycle-specific values relocated into the COLR will continue to be controlled by the BVPS programs and procedures. Each accident analysis addressed in the UFSAR will be examined with respect to changes in the cycle-dependent parameters, which are obtained from the use of NRC approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be conducted per the requirements of 10 CFR 50.59, will ensure that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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 amendment is a programmatic and administrative change and does not result in any change in the manner in which the plant is operated or the way in which the Reactor Protection System provides plant protection. All of the accident transients analyzed in the UFSAR will continue to be protected by the same trip functions with the required trip setpoints. Removal of the cycle specific variables has no influence or impact on, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The cycle specific variables are calculated using the NRC approved methods, and submitted to the NRC to allow the staff to continue to review the values of these limits. The technical specifications will continue to require operation within the core operating limits, and appropriate actions will be required if these limits are exceeded.

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

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

The proposed change to relocate core safety limits, trip setpoint parameter values, and DNB parameter values to the Core Operating Limits Report represents an administrative change and no hardware changes are involved; therefore, no accident analysis acceptance criteria are affected.

The margin of safety is not affected by the removal of cycle-specific core operating limits from the technical specifications. The margin of safety presently provided by current technical specifications remains unchanged. Appropriate measures exist to control the values of these cycle specific limits. The proposed amendment continues to require operation within the core limits as obtained from NRC-approved methodologies, and the actions to be taken if a limit is exceeded will continue to require that the plant be placed in Hot Standby within one hour.

The development of the limits for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload will involve a 10 CFR 50.59 safety review to assure that operation of the unit within the cycle-specific limits will not involve a significant reduction in the margin of safety.

The proposed amendment is a programmatic and administrative change that provides assurance that plant operations continue to be conducted in a safe manner. The proposed amendment does not result in any change in the manner in which the plant is operated or the way in which the Reactor Protection System (RPS) provides plant protection. The proposed relocation does not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. Therefore, the response of the RPS to accident transients described in the UFSAR is unaffected by this change.

As stated previously, this portion of the proposed amendment does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions.

The accident transients are unaffected and the safety analysis acceptance limits are unaffected. The design of the facility and system operating parameters are not being changed.

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

E. Relocation to Licensing Requirements Manual (LRM)

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

The proposed change does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any event initiator. Nor is any analyzed accident scenario being revised. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

The proposed change also does not involve a significant increase in the consequences of an accident previously evaluated. The change does not reduce the operability requirements for the affected instrumentation. The proposed relocation of TS requirements only affects the level of regulatory control involved in future changes to the requirements. The instrument setpoints will continue to be maintained in a similar manner as before. The conclusions and descriptions of the safety analyses described in the UFSAR remain unchanged.

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 does not involve any physical changes to the plant or the modes of plant operation defined in the technical specifications. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

There are no changes in this amendment which would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance requirements are imposed. The proposed amendment has no effect on any previously evaluated accident.

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

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

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed change does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed change does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this change revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed change does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The relocation of TS requirements does not reduce the effectiveness of the requirements being relocated. Rather, the relocation of the TS requirements results in a change in the regulatory control required for future changes made to the requirements. The relocated requirements will continue to be implemented by the appropriate plant procedures (e.g., operating and maintenance procedures) in the same manner as before. However, future changes to the relocated requirements will be controlled in accordance with 10 CFR 50.59 instead of 10 CFR 50.90. The provisions of 10 CFR 50.59 establish adequate controls over requirements removed from the TS and assure future changes to these requirements will be consistent with safe plant operation.

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

F. Miscellaneous Changes

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

The administrative change, for BVPS Unit No. 1 only, pertaining to two loop operation and Reactor Coolant System isolation valve position, does not affect plant safety. The technical specification requirements in LCOs 3.4.1.1 and 3.4.1.4.1 will continue to prohibit two loop operation and ensure safe plant operation by properly controlling the operation and position of the reactor coolant loops and Reactor Coolant System isolation valves.

The administrative change to delete line item 7.d, pertaining to Auxiliary Feedwater (AFW) Pump Auto-start on Emergency Bus Undervoltage, for BVPS Unit No. 1 only, from TS Tables 3.3–3, 3.3–4, and 4.3–2 will not affect plant safety because this function is not directly initiated by bus undervoltage. Rather, the automatic start of the motor-driven AFW pumps is accomplished by the combination of 1) Emergency Bus feed breaker opening 2) valid start signal from ESFAS, and 3) Emergency Diesel Generator (EDG) sequencer actuation. Requirements for these items are included in the ESFAS related TS, Table 3.3–3 and 3.3–4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b). Therefore, since there is no change made to the plant hardware or its operation and requirements related to the AFW pump auto-start function are maintained elsewhere in the BVPS Unit No. 1 TS, deleting line item 7.d from BVPS Unit No. 1 TS Tables 3.3–3, 3.3–4, and 4.3–2 will not significantly change the probability or consequences of any accident previously evaluated.

The proposed change does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any event initiator. Nor is any analyzed accident scenario being revised. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

The proposed change also does not involve a significant increase in the consequences of an accident previously evaluated. The change does not reduce the effectiveness or scope of the affected TS. The proposed changes are administrative in nature and do not affect any technical or equipment operability requirements. The conclusions and descriptions of the safety analyses described in the UFSAR remain unchanged.

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 does not involve any physical changes to the plant or the modes of plant operation defined in the TS. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

There are no changes in this amendment which would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance requirements are imposed. The proposed amendment has no effect on any previously evaluated accident.

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

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

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed change does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed change does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this change revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed change does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The administrative change to delete line item 7.d, pertaining to AFW Pump Auto-start on Emergency Bus Undervoltage, BVPS Unit No. 1 only, from TS Tables 3.3–3, 3.3–4, and 4.3–2 will not affect plant safety because this function is not directly initiated by bus undervoltage. Rather, the automatic start of the motor-driven AFW pumps is accomplished by the combination of (1) Emergency Bus feed breaker opening, (2) valid start signal from ESFAS, and (3) EDG sequencer actuation. Requirements for these items are included in the ESFAS related TS, Table 3.3–3 and 3.3–4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b). Therefore, since there is no change made to the plant hardware or its operation and requirements related to the AFW pump auto-start function are maintained elsewhere in the BVPS Unit No.1 TS, deleting line item 7.d from BVPS Unit No. 1 TS Tables 3.3–3, 3.3–4, and 4.3–2 will not involve a significant reduction in a margin of safety.

The administrative change, for BVPS Unit No. 1 only, pertaining to two loop operation and Reactor Coolant System isolation valve position, does not affect plant safety. The technical specification requirements in LCOs 3.4.1.1 and 3.4.1.4.1 will continue to prohibit two-loop operation and ensure safe plant operation by properly controlling the operation and position of the reactor coolant loops and Reactor Coolant System isolation valve.

The other proposed changes are also administrative in nature and only affect the format or presentation of information in the TS. The proposed changes have no affect on the conclusions or descriptions of the safety analyses described in the UFSAR.

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

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: March 12, 2001.

Description of amendment request: The proposed amendments would revise the current 72-hour allowed outage time (AOT) specified in Technical Specification (TS) 3.8.1.1, Actions ''b'' and ''f,'' and associated TSs 3.4.3 and 3.5.2, to allow 14 days to restore an inoperable emergency diesel generator (EDG) to operable status. The proposed AOT is based on the licensee's integrated assessment of plant operations, deterministic design basis factors, and an evaluation of overall plant risk using probabilistic safety assessment techniques. Additionally, the proposed amendments would relocate TS Surveillance Requirement 4.8.1.1.2.g.1 to a licensee controlled maintenance program that will be incorporated by reference into the Updated Final Safety Analysis Report (UFSAR).

The proposed amendments would also make administrative changes that consist of deleting footnotes on pages 3/4 4–9, 3/4 5–4, 3/4 8–2, and 3/4 8–4, that are no longer applicable, and adding appropriate footnotes on pages 3/4 4–9 and 3/4 8–2 that are compatible with the revised EDG AOT.

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 amendments for Turkey Point Unit 3 and Unit 4 will extend the AOT for a single inoperable EDG from 72 hours to 14 days. The EDGs are designed as backup AC power sources for essential safety systems in the event of a loss of offsite power. As such, the EDGs are not accident initiators, and an extended AOT to restore operability of an inoperable diesel generator would not significantly increase the probability of occurrence of accidents previously analyzed.

The proposed Technical Specification revisions involve the AOT for a single inoperable EDG, and do not change the conditions, operating configuration, or minimum amount of operating equipment assumed in the plant safety analyses for accident mitigation. Plant defense-in-depth capabilities will be maintained with the proposed AOT, and the design basis for electric power systems will continue to conform with 10 CFR 50, Appendix A, General Design Criterion 17. In addition, a Probability Safety Assessment (PSA) was performed to quantitatively assess the risk-impact of the proposed amendment for each unit. The impact on the early radiological release probability for design basis events was also evaluated and it is concluded that the risk contribution from this proposed AOT is small and consistent with regulatory risk-assessment acceptance guidelines.

The relocation of the TS Surveillance requirement 4.8.1.1.2.g.1 from the Technical Specifications to a licensee controlled maintenance program referenced in the UFSAR is bounded by the risk assessment for the EDG AOT extension and therefore does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Therefore, facility operation 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.

The proposed amendments will not change the physical plant or the modes of operation defined in either facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of Turkey Point plant systems. Therefore, facility operation 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 proposed amendments are designed to improve EDG reliability by providing flexibility in the scheduling and performance of preventive and corrective maintenance activities. The proposed changes do not alter the basis for any Technical Specification that is related to the establishment of, or the maintenance of, a nuclear safety margin, and design defense-in-depth capabilities are maintained. The relocation of the TS Surveillance requirement 4.8.1.1.2.g.1 from the Technical Specifications to a licensee controlled maintenance program referenced in the UFSAR is bounded by the risk assessment for the EDG AOT extension. An integrated assessment of the risk impact of extending the AOT for a single inoperable EDG has determined that the risk contribution is small and is within regulatory guidelines for an acceptable TS change. Therefore, facility operation in accordance with the proposed amendments 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.

The proposed changes which consist of deleting four footnotes that are no longer applicable, and adding two footnotes that are compatible with the revised EDG AOT are administrative in nature. Therefore, the staff also proposes to determine that these proposed changes involve no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: March 21, 2001.

Description of amendment request: The licensee proposes to implement a repair roll (re-roll) process for the Crystal River Unit 3 (CR–3) Once Through Steam Generator (OTSG) tubes applicable to the upper and lower tubesheets.

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.

The re-roll process is a method to create a new primary-to-secondary pressure boundary joint in the upper tubesheet of Babcock & Wilcox (B&W) Once Through Steam Generators (OTSGs) manufactured with Inconel Alloy 600 tubes. The new pressure boundary is established by the re-roll to remove degradation of the existing roll joint from pressure boundary service. The re-roll process has been previously qualified as an acceptable repair methodology for use in the upper tubesheet of the Crystal River Unit 3 (CR–3) OTSGs by License Amendment No. 180. This proposed LAR incorporates Revision 4 of Topical Report BAW–2303P, ‘‘OTSG Repair Roll Qualification Report.’’ This proposed LAR also addresses several editorial changes which do not impact the current CR–3 accident analyses.

The qualification of the OTSG tube re-roll methodology is based on establishing a mechanical joint length that will carry all structural loads imposed on the OTSG tubes while maintaining the required margins during normal and accident conditions. A series of tests and analyses were performed to establish the minimum acceptable length of the OTSG tube re-roll. Tests performed included leak, tensile, fatigue, ultimate load and eddy-current measurement uncertainty. The analyses evaluated plant operating and faulted load conditions. OTSG tube leakage remains bounded by the evaluation presented in the CR–3 Final Safety Analysis Report (FSAR) for a main steam line break (MSLB). The current CR–3 Improved Technical Specifications (ITS) include a description of the required inspection program for the OTSG tube re-rolls. The required ITS inspections following OTSG tube re-roll installation, and during future inservice inspections, ensure continuous monitoring of these tubes such that in service degradation of tubes repaired by the re-roll process will be detected. Based on the qualification testing and analyses performed, as well as the industry experience with the use of the OTSG tube re-roll processes, there are no new safety issues associated with the use of the re-roll methodology. The probability of a steam generator tube rupture is not increased by the re-roll since it is a repair process not applied to defective OTSG tube areas. This repair process establishes a new pressure boundary roll joint which is free of degradation. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The re-roll process creates no new failure modes or accident scenarios. The new pressure boundary joint created by the re-roll process has been demonstrated, by testing and analysis, to provide structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. Furthermore, testing and analysis demonstrate that the re-roll process creates no new adverse effects for the repaired tube and does not change the design or operating characteristics of the OTSGs. BAW–2303P, Revision 4, addresses limiting events for steam generator re-roll repairs. These events include Main Steam Line Break, Small Break Loss of Coolant Accident and other transients on the B&W Once Through Steam Generators. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The re-roll process effectively removes the defective/degraded area of the tube from service by establishing a new pressure boundary. The re-roll interface created with the tubesheet satisfies the necessary structural, leakage and heat transfer requirements. Implementation of BAW–2303P, Revision 4, will result in assurance that parameters affecting the integrity of steam generator tubes continue to meet safety analyses and industry codes and standards. Therefore, the FSAR analyzed accident scenarios remain bounding, and the use of the re-roll process does not significantly reduce the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: March 28, 2001.

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 Improved Technical Specifications (ITS) 3.7.18, ‘‘Control Complex Cooling System’’ to allow a one-time increase in the Completion Time for restoring an inoperable Control Complex Cooling System train from 7 days to 35 days.

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

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

The Control Complex Cooling System is not an initiator of any design basis accident. The Control Complex Cooling System is designed to provide sufficient cooling to ensure operability of safety-related equipment located in the control room and other portions of the control complex under normal and accident conditions.

The proposed license amendment extends the Completion Time for restoring an inoperable Control Complex Cooling train from 7 days to 35 days on a one-time basis for each train to allow on-line refurbishment of the control complex chillers. The proposed amendment also specifies that the requirements of (LCO) 3.0.4 are not applicable to ITS 3.7.18 Condition A during the 35-day Completion Times. The design functions of the Control Complex Cooling System and the initial conditions for accidents that require the Control Complex Cooling System will not be affected by the change. The increased Completion Time requested by License Amendment Request (LAR) #259 results in slight increases in core damage frequency and core damage probability; however, these increases are well below values that are considered risk significant. Therefore, the change will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed amendment extends the Completion Time for restoring an inoperable Control Complex Cooling System train on a one-time basis for each train to allow on-line performance of maintenance activities that will improve chiller reliability. The proposed amendment will not result in changes to the design, physical configuration or operation of the plant. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed license amendment increases the Completion Time for restoring an inoperable Control Complex Cooling System train from 7 days to 35 days on a one-time basis for each train. The proposed amendment also specifies that the requirements of Limiting Condition for Operation (LCO) 3.0.4 are not applicable to ITS 3.7.18 Condition A during the one-time 35-day Completion Times. The proposed changes will maintain operational flexibility while allowing on-line refurbishment of the control complex chillers to improve their reliability and extend their useful lifetimes, thus increasing the long-term margin of safety of the system.

The Control Complex Cooling System is designed to provide sufficient cooling to ensure operability of safety-related equipment located in the control room and other portions of the control complex under both normal and accident conditions. Either redundant train of the system is capable of performing this function; therefore, as long as one train is available, the margin of safety is maintained. Waiving the requirements of LCO 3.0.4 while the requested 35-day Completion Times are in effect will not impact the availability of the redundant system train, backup systems, or required support systems. In addition, since the heat removal requirements for the control room and other vital heat loads in the control complex are the same in Mode 1 as they are in Mode 3, allowing the plant to escalate Modes while chiller repairs are in progress will not impact the ability of the Control Complex Cooling System to fulfill its intended safety function. During the time that the required maintenance activities are being performed on each chiller, the availability of redundant system components will be maximized by administratively controlling preventive maintenance and surveillance activities performed on the Control Complex Cooling System and required support systems. Defense-in-depth measures will also be implemented to ensure the availability of temporary and permanently installed backup systems capable of providing cooling to the control room and the other vital equipment areas in the control complex. Although the increased Completion Time requested by LAR #259 results in a loss of redundancy and slight increases in core damage frequency and core damage probability, these increases are well below values that are considered risk significant. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: February 28, 2001.

Description of amendment request: The proposed amendment changes the Seabrook Station Technical Specifications (TSs) 3/4.8.1.1 A.C. Sources—Operating. In addition, other changes are proposed either for clarity, which are reflective of the improved Standard Technical Specifications for Westinghouse Plants, NUREG–1431, Rev. 1 and Draft Rev. 2, or do not meet the four criteria of 10 CFR 50.36 for inclusion in TSs. Those requirements that do not meet the criteria for inclusion in the TSs will either be deleted or relocated to the Seabrook Station Technical Requirements manual.

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

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

The proposed changes do not involve a change in the operational limits, do not involve a change in physical design of the electrical power systems, do not change the function or operation of plant equipment or affect the response of that equipment if called upon to operate. The proposed allowed outage time extensions will not cause a significant increase in the probability or consequences of an accident previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of accidents previously evaluated.

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

No new or different kind of accident is created because the proposed changes do not involve a change in the operational limits, do not involve a change in physical design of the electrical power systems, do not change the function or operation of plant equipment or introduce any new failure mechanisms. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes. 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 the margin of safety.

The margin of safety will remain the same because the proposed changes do not involve a change in the operational limits, do not involve a change in physical design of the electrical power systems, do not change the function or operation of plant equipment or affect the response of that equipment if it is called upon to operate. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 21, 2000.

Description of amendment request: The proposed amendment would allow plant operation to continue if the temperature of the Ultimate Heat Sink (UHS) exceeds the Technical Specification limit of 75 °F provided the water temperature, averaged over the previous 24-hour period, is at or below 75 °F. The proposed operational flexibility would only apply if the UHS temperature is between 75 °F and 77 °F. The current action time requirements would still apply if the UHS temperature exceeds 77 °F, or if the 24-hour averaged value exceeds 75 °F. The current Technical Specification Limiting Condition for Operation (LCO) limit of 75 °F would not be changed. The Bases for the associated Technical Specification would also be modified.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representations made by the licensee in the December 21, 2000, application, is presented below:

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

The proposed changes will allow plant operation to continue if the temperature of the UHS exceeds the Technical Specification limit of 75 °F provided that: (1) The water temperature, averaged over the previous 24 hour period, is at or below 75 °F, and (2) the UHS temperature is less than or equal to 77 °F. This increase in UHS temperature will not affect the normal operation of the plant to the extent which would make any accident more likely to occur. In addition, there exists adequate margin in the safety systems and heat exchangers to assure the safety functions are met at the higher temperature. An evaluation has confirmed that safe shutdown will be achieved and maintained for a loss of coolant accident (LOCA) with a loss of normal power (LNP) and a single active failure with an UHS water temperature as high as 77 °F.

Thus, the proposed changes will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed changes can not cause an accident. Therefore, there will be no increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will allow plant operation to continue if the temperature of the UHS exceeds the Technical Specification limit of 75 °F provided that: (1) The water temperature, averaged over the previous 24 hour period, is at or below 75 °F, and (2) the UHS temperature is less than or equal to 77 °F. This will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The proposed changes do not introduce any new failure modes. Also, the response of the plant and the operators following these accidents is unaffected by the changes. In addition, the UHS is not an accident initiator. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes will allow plant operation to continue if the temperature of the UHS exceeds the Technical Specification limit of 75 °F provided that: (1) The water temperature, averaged over the previous 24 hour period, is at or below 75 °F, and (2) the UHS temperature is less than or equal to 77 °F. The licensee performed an evaluation of the safety systems to ensure their safety functions can be met with a UHS water temperature of 77 °F. The evaluation determined that an increase in UHS temperature from 75 °F to 77 °F would nominally cause a 2 °F temperature increase in service water system, reactor building closed cooling water system, and associated heat exchanger loads. This represents a slight reduction in the margins of safety in terms of these systems' abilities to remove accident heat loads, and in terms of the thermally induced pipe stresses within these systems during accident conditions. As part of its evaluation, however, the licensee verified that these safety systems will still be able to perform their design basis functions, and that pipe stresses will remain within allowable levels.

Safe shutdown capability has been demonstrated for a UHS water temperature as high as 77 °F.

The proposed changes will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: January 5, 2001.

Description of amendment request: The proposed change would amend the Salem Nuclear Generating Station, Unit Nos. 1 and 2 Technical Specifications (TSs) by adding a requirement to perform a Hydrogen Analyzer gas calibration at least once per 92 days, and changing the required frequency to perform a channel calibration of the Hydrogen Analyzer from once per 92 days to once per refueling.

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

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

The Hydrogen Analyzer provides detection and measurement of containment hydrogen concentration so that hydrogen concentration can be maintained below its flammable limit following a Loss of Coolant Accident. As such the Hydrogen Analyzer does not affect the probability of any previously evaluated accident.

The proposed changes are consistent with the manufacturer's recommendations to ensure that the Hydrogen Analyzer will provide accurate indication of containment hydrogen concentration when required. Under the proposed change, a gas calibration consisting of all elements of the Hydrogen Analyzer channel calibration, with the exception of the calibration of the instrument's resistance temperature detector and pressure transducer, would be performed at least every 92 days. As a part of the gas calibration, a comparison of the indication of Hydrogen Analyzer resistance temperature detector and the pressure transducer against installed plant instrumentation measuring containment temperature and pressure would be performed. At least once per each refueling, a channel calibration of the Hydrogen Analyzers, including a calibration of the instrument's resistance temperature detector and pressure transducer using a secondary standard of a specified accuracy would be performed. Therefore, the proposed change would not affect the consequences of any previously evaluated accident.

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

The proposed change affects only the specified calibration frequency of the Hydrogen Analyzers. The proposed surveillance frequency complies with the manufacturer's recommendations and will ensure that the Hydrogen Analyzers will provide accurate indication of containment hydrogen concentration when required. The change will not affect the design of any Salem Generating Station structure, system, or component, nor would it result in any new plant configuration. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from [any] accident previously evaluated.

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

The proposed change to the Hydrogen Analyzer calibration frequency will not affect the design or operating limits of any Salem

Generating Station structure, system, or component. The proposed surveillance frequency complies with the manufacturer's recommendations and will ensure that the Hydrogen Analyzers will provide accurate indication of containment hydrogen concentration when required. Therefore the proposed changes to the Technical Specifications do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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: December 28, 2000.

Description of amendment request: This proposed amendment would revise Technical Specification (TS) 3/4.3.2, and Tables 3.3–3 and 3.3–4 to incorporate consistent applicability and action for Engineered Safety Feature Actuation System (ESFAS) Instrumentation, Functional Unit 5.b. (Automatic Actuation Logic and Actuation Relay) Turbine Trip and Feedwater Isolation. This change will provide consistency between Tables 3.3–3, 3.3–4, and 4.3–2, and will be similar to the equivalent requirement in NUREG–1431, Revision 1.

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

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

The addition of an ACTION STATEMENT and the addition of an AOT [allowed outage time] (and its associated actions if not met) for a TS action statement are neither an accident initiator or precursor. The ESFAS actuates in response to an accident and has a mitigating function. Increasing the TS requirements for specific TS instrument loops provides additional assurance that the channels will be capable of performing their design function in the event of a DBA [design-basis accident]. The ability of the operations staff to respond to an evaluated accident or plant transient will not be hampered. This change provides conservative requirements to assure that the design basis of the plant is maintained.

Addition of conservative changes to the Engineered Safety Feature Actuation System Instrumentation do not contribute to the initiation of any accident evaluated in the FSAR [Final Safety Analysis Report]. Supporting factors are as follows:

- The changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. These changes are conservative and consistent with the Standard Technical Specifications, NUREG–1431, Rev. 1. There are no deletions from the Technical Specifications made by these changes, nor relaxation in any applicability, action, or surveillance requirements.
- Overall plant performance and operation is not altered by the proposed changes. There are to be no plant hardware changes as a result of this proposed change and only minimal procedural changes.

Therefore, since the Engineered Safety Feature Actuation System Instrumentation are treated more conservatively, the probability of occurrence or consequences of an accident evaluated in the VCSNS FSAR will be no greater than the original design basis of the plant.

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

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

The proposed changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. Additionally, the addition of an ACTION STATEMENT and an AOT with conservative requirements are intended to assure that the plant is in a safe configuration and can meet accident analyses assumptions. These changes are conservative and consistent with the Improved Technical Specifications, NUREG–1431, Rev. 1. No new accident initiator mechanisms are introduced since:

- No physical changes to the Engineered Safety Feature Actuation System Instrumentation are made.
- No deletions from the Technical Specifications are made.
- No relaxation in any applicability, action, or surveillance requirements are made.

Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

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

The proposed change requires that an instrument channel for an Engineered Safety Feature remain operable or be restored to operability within a reasonable time period, otherwise a controlled shutdown is required. This conforms to the safety analysis where the plant and its systems, structures and components must be capable of performing the safety function while a DBA is occurring, in the presence of a worst case single failure.

This is not a reduction in a margin of safety, since it restores the margin that was designed into the plant.

The proposed changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. These changes are conservative and consistent with the Standard Technical Specifications, NUREG–0452, Rev. 5.

The proposed changes impose more restrictive operating limitations, and their use provides increased assurance that the Engineered Safety Feature Actuation System Instrumentation remains operable. Since the changes are conservative additions, it is concluded that the changes do not involve a significant reduction in the margin of safety. This is not a reduction in a margin of safety, since it restores the margin that was designed into the plant.

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.

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

Date of amendment requests: March 21, 2001.

Description of amendment requests: The amendment application proposes to revise the Facility Operating License No. NPF–10, and Facility Operating License No. NPF–15 for San Onofre Nuclear Generating Station, Units 2 and 3, respectively. The licensee proposed to simplify the Facility Operating Licenses by deleting those license conditions which have been completed and are no longer required to be identified in the licenses.

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 any accident previously evaluated?

Response: No.

This proposed change is administrative since it only deletes completed San Onofre Units 2 and 3 license conditions, providing appropriate references and discussion of the actions taken which document their completion. There is no physical plant change or change to plant operation which could increase the probability or consequences of any accident previously evaluated.

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

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

Response: No.

This proposed change is administrative because it only deletes completed Onofre Units 2 and 3 license conditions and there is no physical plant change or change to plant operation which could introduce any mechanism which could create a new or different kind of accident.

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

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

Response: No.

This change is administrative because it only deletes completed San Onofre Units 2 and 3 license conditions and there is no physical plant change or change to plant operation, therefore there is no impact in a margin of safety.

Therefore, a significant reduction in a 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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: March 21, 2001.

Description of amendment requests: The amendment application proposes to revise the San Onofre Nuclear Generating Station, Units 2 and 3, Technical Specification (TS) to clarify the methodology used to test the Control Room Emergency Air Cleanup System and Post-Accident Cleanup Filter System High Efficiency Particulate Air (HEPA) filters. Specifically, in TS 5.5.2.12, ''Ventilation Filter Testing Program (VFTP),'' the reference to the American Society for Mechanical Engineers (ASME) Code ASME N510–1989 will be revised to the American National Standards Institute (ANSI) N510–1975. Also, in TS 5.5.2.12.d, references to Regulatory Guide (RG) 1.52, Revision 2, and ASME N510–1989 will be deleted.

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

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

Response: No.

The proposed change is to change the reference to ASME Code in subsections 5.5.2.12.a and 5.5.2.12.b from ASME N510–1989 to ASME N510–1975. Technical Specification (TS) 3.7.11, ''Control Room Emergency Air Cleanup System (CREACUS)'' Surveillance Requirement (SR) 3.7.11.2 and TS 3.7.14, ''Fuel Handling Building Post-Accident Cleanup Filter System (PACU),'' SR 3.7.14.2 require CREACUS and PACU filter testing ''in accordance with the Ventilation Filter Testing Program (VFTP).''

San Onofre Nuclear Generating Station (SONGS) TS 5.5.2.12.a, ''Ventilation Filter Testing Program,'' states that the in-place HEPA filter testing is performed in accordance with Regulatory Guide (RG) 1.52, Revision 2 and ASME N510–1989. However, the CREACUS in-place HEPA filter testing uses a method (''Alternate Shroud Test'') which is no longer specified in ASME N510–1989. But this method is specified in ANSI N510–1975 and was used when the plant was licensed. In addition, the PACU in-place HEPA filter testing methodology which is employed at SONGS has a downstream point location which differs from the location suggested in ASME N510–1989. ANSI N510–1975, while providing a suggestion where downstream sample could be located, nevertheless does not provide a specific location. The test acceptance criteria are the same for methods cited in ANSI N510–1975 and ASME N510–1989. The method which is employed at SONGS provides more conservative results because the test is performed on individual HEPA filters, which ensures that each of the HEPA filters in the tested bank meets the acceptance criteria.

The locations of the PACU HEPA downstream sample points are different from the location suggested in ASME N510–1989, though they meet the requirements delineated in ANSI N510–1975. ANSI N510–1975 requires that a single representative downstream sample point be established, if possible, at the location where adequate mixing may be achieved, or at a point downstream of a fan, or multiple downstream sampling points may be used (such as in the Alternate Shroud Technique used in the CREACUS system) if a single downstream sample point is not feasible.

Since the HEPA filters are tested to the same acceptance criteria, and the testing methodology is permitted by ANSI N510–1975, to which the plant is licensed, it is concluded that the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Section 5.5.12.d will be modified by the proposed change by deleting the references to RG 1.52, Revision 2 and ASME N510–1989. There are no requirements for pressure drop test across combined HEPA filters, the prefilters, and the charcoal absorbers in RG 1.52, Revision 2, and ASME N510–1989. The proposed version of section 5.5.2.12.d reads:

''Testing to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal absorbers, when tested at the appropriate system flowrate.''

The proposed change clarifies the statement of section 5.5.2.12.d. Pressure drop testing across combined HEPA filters, the prefilters, and the charcoal adsorbers is industry-wide practice which is based on good engineering practice and operating experience.

Therefore, the probability or consequences of an accident previously evaluated will not be increased by operating the facility in accordance with this proposed change.

(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?

Response: No.

The proposed change does not change the design or configuration of the plant. The proposed change is to change the reference to ASME Code in subsection 5.5.2.12.a and 5.5.2.12.b from ASME N510–1989 to ANSI N510–1975 to more clearly reflect the standard used. Also, subsection 5.5.2.12.d will be changed by deleting the references to RG 1.52, Revision 2, and ASME N510–1989 regarding pressure drop test across HEPA filters. RG 1.52, Revision 2, and ASME N510–1989 do not require pressure drop test across HEPA filters.

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

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

Response: No.

The proposed change is to change the reference to the ASME Code in subsections 5.5.2.12.a and 5.5.2.12.b from ASME N510–1989 to ANSI N510–1975. The CREACUS units' HEPA filters are currently tested to ANSI N510–1975. Although the test methodology is slightly different than that in ASME N510–1989, the acceptance criteria are the same and the current methodology is conservative. Thus the current testing satisfies the acceptance criteria of ASME N510–1989, even though the test method is different.

The current methodology for HEPA filter testing will not change as a result of the proposed change. Also, deletion of reference to RG 1.52, Revision 2, and ASME N510–1989 from subsection 5.5.2.12.d clarifies this section because these standards do not require HEPA filters pressure drop test. Consequently, there is no change to the design or operation of the plant as a result of this change.

Therefore, operation of the facility in accordance with this 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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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

Date of application for amendments: March 12, 2001 (TS 99–18).

Brief description of amendments: The proposed amendment would revise the Sequoyah Nuclear Plant (SQN) Technical Specifications (TSs). The revision would delete TS 4.7.7.a and add proposed TS 3/4.7.13 regarding the control room air conditioning system to make the SQN TSs more consistent with the Westinghouse Standard TSs.

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.

TVA has identified Surveillance Requirement (SR) 4.7.7.a, which determines operability of the main control room emergency ventilation system (CREVS) relative to temperature, to be inadequate and nonconservative. TVA proposes to deleted this SR coincident with the addition of a new TS 3/4.7.13. The proposed TS addition for the main control room air-conditioning system (CRACS) provides a more adequate SR for determination of operability with associated actions to take for inoperability; resolves an inadequate TS in accordance with the guidance in NRC [U.S. Nuclear Regulatory Commission] Administrative Letter 98–10; establishes clarity between CRACS and CREVS; and provides greater consistency with NUREG–1431 and TSTF–51 [TS Task Force issue Traveler No. 51], Revision 2. These proposed revisions are conservative and are not the result of a change to plant equipment, system design, testing methods, or operating practices. Since the proposed revisions will increase conservatism and the systems will continue to meet their required safety function without plant modification or operating practices, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed revisions to the SQN TSs will not alter plant equipment or operating practices. The change will not result in the installation of any new equipment or systems. The intent of deleting the SR and adding a specification is to address a nonconservative TS, provide clarification of plant systems, and improve consistency with NUREG–1431. Since the systems' functions are associated with accident mitigation and will continue to perform without change and were not previously considered to contribute to accident generation, the proposed changes will not create the possibility of a new or different kind of accident.

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

Both the main control room (MCR) emergency ventilation and air-conditioning systems provide for the safe, uninterrupted occupancy of the MCR during an accident and the subsequent recovery period. The proposed TS revisions will not change the methods of operating the plant or setpoints associated with safety-related equipment in the implementation of this request. Therefore, the proposed revisions do not involve a reduction in the margin of safety.

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: Richard P. Correia.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: January 15, 2001.

Brief description of amendment: The amendment revised TS 5.4.2(f) to remove the fuel assembly U–235 loading criterion for fuel assemblies stored in the spent fuel storage pool.

Date of issuance: March 26, 2001.

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance including issuance of approval of changes to the Updated Final Safety Analysis Report as described in the Nuclear Regulatory Commission staff's safety evaluation.

Amendment No.: 231.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11051).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 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.

Brief description of amendment: The amendment revised the frequency for maintenance inspections of the emergency diesel generators from annually to once every 2 years and stated that the inspections shall be conducted in accordance with procedures developed in conjunction with applicable Fairbanks Morse Owners Group and manufacturer's recommendations.

Date of issuance: March 29, 2001.

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

Amendment No.: 232.

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

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2012).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 1, 2000.

Brief description of amendments: The amendments revise the value of the minimum departure from nucleate boiling ratio (DNBR) from ''≥ 1.30'' in the current Technical Specifications (TSs) to ''≥ 1.3 (through operating cycle 10)'' and ''≥ 1.34 (operating cycle 11 and later)'' in the safety limits TS 2.1.1.1 and in function 15, DNBR—Low, in Table 3.3.1–1, ''Reactor Protective System Instrumentation.'' The amendments are structured such that the ''≥ 1.34'' would become effective for each unit in operating cycle 11 and later. Operating cycle 11 begins in spring 2002 for Unit 2, in fall 2002 for Unit 1, and in spring 2003 for Unit 3. From now to operating cycle 11, the ''≥ 1.30'' will remain the minimum DNBR requirement for the three units.

Date of issuance: March 28, 2001

Effective date: March 28, 2001, and shall be implemented within 60 days of the date of issuance.

Amendment Nos.: Unit 1–133, Unit 2–133, Unit 3–133

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7670)

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 5, 2000.

Brief description of amendments: The amendments revise the action statement for Specification 3.7.5, ''Auxiliary Feedwater (AFW) System,'' of the Technical Specifications (TSs). The amendments incorporate NRC-approved TS Task Force (TSTF) Traveler Number TSTF–340, Revision 3, to allow a 7-day completion time for the turbine-driven AFW pump if inoperability occurs in reactor Mode 3 following a refueling outage, and if Mode 2 had not been entered.

Date of issuance: March 29, 2001.

Effective date: March 29, 2001, and shall be implemented within 45 days of the date of issuance.

Amendment Nos.: Unit 1–134, Unit 2–134, Unit 3–134.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 24, 2001.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2001.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: January 25, 2000, as supplemented on October 31, 2000, and December 18, 2000.

Brief description of amendments: The amendments revise the Calvert Cliffs Technical Specifications to eliminate response time testing for those pressure sensors which were discussed and approved in the Combustion Engineering Owners Group Topical Report NPSD–1167.

Date of issuance: March 22, 2001.

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

Amendment Nos.: 244 and 218.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 6403).

The October 31 and December 18, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 22, 2001.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket Nos. 50–32 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: December 1, 2000.

Brief Description of amendments: The amendments change the Technical Specifications for the submittal date of the ''Radioactive Effluent Release Report'' to ''prior to May 1'' of each year.

Date of issuance: March 21, 2001.

Effective date: March 21, 2001.

Amendment Nos.: 212 and 239.

Facility OperatingLicense Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9381).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 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, as revised by letter dated January 12, 2001.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) to allow Type B and C containment leak rate testing to be performed in accordance with 10 CFR Part 50, Appendix J, Option B. The amendment also increases the interval in TS Surveillance Requirement 3.6.2.2 for containment air lock door interlock testing from 18 months to 24 months.

Date of issuance: March 30, 2001.

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

Amendment No.: 194.

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

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7676).

The licensee's letter dated January 12, 2001, did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 30, 2001.

No significant hazards consideration comments received: No.

Duke Energy Corporation, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendment: February 20, 2001.

Brief description of amendment: The amendment revised the Technical Specifications Table 3.3.2–1 for Catawba Nuclear Station Unit 1. It modified the required actions for the Engineered Safety Feature Actuation System Table 3.3.2–1, function 6.f (auxiliary feedwater (AFW), auxiliary feedwater pump train A and train B suction transfer on suction pressure—low) on a one-time basis. The proposed one-time change will require that if more than one channel of low suction pressure instrumentation becomes inoperable, in lieu of requiring unit shutdown within 7 hours, the licensee will immediately enter the applicable condition(s) or required action(s) for the associated AFW train made inoperable by the inoperable channels. This modification will support the timely replacement of a broken pressure switch in the Train B of AFW suction transfer on low suction pressure function.

Date of issuance: April 6, 2001.

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

Amendment No.: 190.

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

Date of initial notice in ***Federal Register:*** February 27, 2001 (66 FR 12568). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 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: November 2, 2000.

Brief description of amendment: The amendment made the following changes: (1) added a new Technical Specification (TS) 3.3.1.3, ''Oscillation Power Range Monitoring (OPRM) Instrumentation,'' (2) revised TS 3.4.1, ''Recirculation Loops Operating,'' to remove monitoring specifications that are no longer needed upon activation of the automatic OPRM instrumentation, and (3) revised TS 5.6.5 to include in the Core Operating Limits Report the applicable operating limits for the OPRM and also reference the topical report that describes the analytical methods used to determine the setpoint values for the OPRM.

Date of issuance: April 5, 2001.

Effective date: April 5, 2001, and shall be implemented prior to restart from Refuel Outage 15.

Amendment No.: 171.

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

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77916).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 1, 2000.

Brief description of amendment: The amendment approves a change to the Pilgrim Technical Specification Table 4.6–3. The change modifies the reactor pressure vessel surveillance capsule withdrawal schedule by substituting ''21 (approx)'' under the column ''Effective Full Power Years (EFPY)'' for the current ''18 (approx).''

Date of issuance: April 2, 2001.

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

Amendment No.: 188.

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

Date of initial notice in ***Federal Register:*** November 1, 2000 (65 FR 65342).

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: September 28, 2000, as supplemented by letters dated October 26, 2000, and February 19, 2001.

Brief description of amendment: The amendment changes the Arkansas Nuclear One, Unit 1 technical specifications to allow a revised reroll repair process for the steam generators.

Date of issuance: March 28, 2001.

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

Amendment No.: 212.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77519).

The supplemental letters dated October 26, 2000, and February 19, 2001, provided additional information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: August 29, 2000, as supplemented by letter dated March 2, 2001.

Brief description of amendment: The amendment revises the TS to allow steam generator tubes to remain in service with indications of outer diameter intergranular attack (ODIGA) in the upper tubesheet region of the steam generators.

Date of issuance: March 28, 2001.

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

Amendment No.: 213.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77917).

The supplemental letter dated March 2, 2001, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: November 20, 2000.

Brief description of amendment: Revised Technical Specification Figure 3.4.6.1–1, which affects heatup, cooldown and inservice test Pressure-Temperature limitations. The revisions are applicable until the end of Operating Cycle 7.

Date of issuance: March 23, 2001.

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

Amendment No.: 111.

Facility Operating License No. NPF–85. This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11058).

The December 20, 2000, letter provided additional information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 18, 2001, as supplemented February 20, and March 26, 2001.

Brief description of amendments: The amendments changed the Technical Specification (TS) Table 1.2, ''Operational Conditions,'' and TS 3/4.9.1, ''Reactor Mode Switch,'' to allow movement of a single control rod with the reactor in hot shutdown or cold shutdown for post-maintenance and surveillance testing of the control rod and the control rod drive. The amendments also changed TS Table 3.3.1–1 to require the nuclear instrumentation system intermediate range monitors (IRMs) to be operable when moving a control rod in hot shutdown or cold shutdown. TS Table 4.3.1.1–1 is changed to add surveillance requirements for the IRMs in hot or cold shutdown.

Date of issuance: April 5, 2001.

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

Amendments Nos.: 149 and 112.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11060). The February 20, and March 26, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 18, 2001.

Brief description of amendments: The amendments changed Technical Specification Surveillance Requirement 4.9.2.d to allow the shorting links to remain in place if adequate shutdown margin has been demonstrated.

Date of issuance: April 5, 2001.

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

Amendments Nos.: 150 and 113.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11059).

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: December 27, 1999.

Brief description of amendments: The amendments revise Technical Specifications (TSs) to increase allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for selected actuation instrumentation. The amendments implement AOT/STI changes based on Topical Reports by General Electric Company and the Boiling Water Reactor Owners' Group which have previously been reviewed and approved by the Nuclear Regulatory Commission (NRC).

Date of issuance: March 28, 2001.

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

Amendment Nos.: 198 and 194.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 9, 2000 (65 FR 48746).

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

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station (BVPS), Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: May 12, 2000, as supplemented by letters dated June 19, November 2, and December 1, 2000, and January 29, 2001.

Brief description of amendments: This amendment authorized changes to the BVPS–1 and 2, Updated Final Safety Analysis Reports (UFSARs) with regard to selected design-basis accident dose consequence calculations. For BVPS–1, changes involve the following DBAs: loss of offsite alternating-current (AC) power, fuel-handling accident, accidental release of waste gas, steam generator tube rupture, rod cluster control assembly ejection, single reactor coolant pump locked rotor, and loss of reactor coolant for small ruptured pipes/loss-of-coolant accidents. For BVPS–2, changes involve the following DBAs: steam system piping failures (or main steam line break), loss of AC power, reactor coolant pump shaft seizure, rod cluster control assembly ejection, failure of small lines carrying primary coolant outside containment, steam generator tube rupture, loss-of-coolant accidents, and waste gas system failure.

Date of issuance: March 22, 2001.

Effective date: This license amendment is effective as of the date of its issuance and shall be implemented by the next update to the UFSAR as required by 10 CFR 50.71(e). Implementation of the amendment requires the incorporation in the UFSAR of the changes to the description of the facility as described in the licensee's application dated May 12, 2000, as supplemented June 19, November 2, and December 1, 2000, and January 29, 2001.

Amendment Nos.: 237 and 119

Facility Operating License Nos. DPR–66 and NPF–73: Amendments authorize revisions to the BVPS–1 and 2 UFSARs.

Date of initial notice in ***Federal Register:*** September 6, 2000 (65 FR 54086).

The June 19, 2000, letter revised the licensee's no significant hazards evaluation and was used, with the original submittal, as a basis for the staff's proposed no significant hazards consideration determination which was published on September 6, 2000. The November 2, and December 1, 2000, and January 29, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice.

The Commission's related evaluation of the UFSAR changes is contained in a Safety Evaluation dated March 22, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 27, 2000, as supplemented March 12, 2001.

Brief description of amendment: The amendments delete Technical Specifications Section 6.8.4.3, ''Post-Accident Sampling,'' thereby eliminating the requirements to have and maintain the post-accident sampling system.

Date of Issuance: March 27, 2001.

Effective Date: March 27, 2001.

Amendment No.: 174 and 114.

Facility Operating License Nos. DPR–67 and NPF–16: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81921). The March 12, 2001, supplement did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 27, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 31, 2000.

Brief description of amendment: This amendment revised Technical Specification (TS) and associated Bases pages Table 3.3.18–1, ''Remote Shutdown System Instrumentation.'' The list of instruments that would be used by operators to place and maintain the plant in a safe shutdown condition from outside the control room have been modified consistent with recent plant modifications and changes to the approach to achieve and maintain a safe shutdown condition.

Date of issuance: March 22, 2001.

Effective date: Date of issuance, to be implemented prior to Fall 2001 Restart.

Amendment No.: 196.

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

Date of initial notice in ***Federal Register:*** October 4, 2000 (65 FR 59223).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 22, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 15, 2000, as supplemented March 7, 2001.

Brief description of amendments: The proposed amendment would revise Technical Specification (TS) 3.2.6, ''Allowable Power Level—APL,'' and TS 1.38, ''Allowable Power Level (APL),'' definitions of APL to make them consistent throughout the TSs.

Date of issuance: March 29, 2001.

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

Amendment Nos.: 251, 233.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81924). The supplement contained clarifying information and did not change the initial no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 29, 2001.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: January 2, 2001, as supplemented March 5, 2001.

Brief description of amendment: The amendment revises Technical Specifications (TS) 3/4.6.2.2.a for the Unit 1 spray additive tank to require a contained volume between 4000 and 4600 gallons of between 30 and 34 percent by weight sodium hydroxide (NaOH) solution. In addition, the amendment makes four types of format changes to the TS pages for Unit 1.

Date of issuance: March 29, 2001.

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

Amendment No.: 252.

Facility Operating License No. DPR–58: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7681).

The March 5, 2001, supplemental letter did not change the scope of the proposed action and did not change the NRC's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2001.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendment: January 13, 2000 (submitted by PP&L, Inc., the licensee before July 1, 2000), as supplemented September 6, 2000 (submitted by PPL Susquehanna, LLC, the licensee on and after July 1, 2000).

Brief description of amendment: The amendments made administrative changes to the Technical Specifications correcting the wording of the legends in Figure 3.4.10.1, ''Reactor Vessel Pressure vs. Minimum Vessel Temperature,'' for both units, and correcting administrative errors in Section 5.6.5.b, regarding the Core Operating Limits Report, for Unit 2.

Date of issuance: March 30, 2001.

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

Amendment Nos.: 192 and 167.

Facility Operating License Nos. NPF–14 and NPF–22: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17918).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated March 30, 2001.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: August 31, 2000, as supplemented October 12 and November 8, 2000, and February 16, 2001.

Brief description of amendment: This amendment conforms the license to reflect the transfer of Operating License No. DPR–21 for the Millstone Nuclear Power Station, Unit No. 1 to the extent held by the Selling Owners to Dominion Nuclear Connecticut, Inc., as previously approved by an Order dated March 9, 2001.

Date of issuance: March 31, 2001.

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

Amendment No.: 109.

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

Date of initial notice in ***Federal Register:*** October 24, 2000 (65 FR 63630).

The October 12 and November 8, 2000, and February 16, 2001 supplements provided clarifying information and did not expand the scope of the application as originally published.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 31, 2000, as supplemented January 4, 2001.

Brief description of amendment: This amendment authorizes changes to the Millstone Nuclear Power Station, Unit No. 2 Final Safety Analysis Report (FSAR) to allow the use of the Siemens Power Corporation U.S. Nuclear Regulatory Commission-approved methodology for determining the fuel centerline melt linear heat rate limit (FCMLHRL) on a cycle-by-cycle basis. Northeast Nuclear Energy Company evaluated this method of calculating FCMLHRL utilizing the criteria of 10 CFR 50.59 and determined that this change required NRC approval before implementation. Technical Specification Bases 2.1.1, ''Reactor Core,'' has also been revised accordingly.

Date of issuance: March 29, 2001.

Effective date: As of the date of issuance and shall be implemented no later than the date of submission of the next update of the FSAR.

Amendment No.: 255.

Facility Operating License No. DPR–65: Amendment authorized changes to the FSAR and revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7684).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2001.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of application for amendment: August 31, 2000, as supplemented October 12 and November 8, 2000, and February 16, 2001.

Brief description of amendment: These amendments conform the licenses to reflect the transfer of Operating Licenses Nos. DPR–65 and NPF–49 for the Millstone Nuclear Power Station, Unit Nos. 2 and 3, to the extent held by the Selling Owners to Dominion Nuclear Connecticut, Inc., as previously approved by an Order dated March 9, 2001.

Date of issuance: March 31, 2001.

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

Amendment Nos.: 256 and 196.

Facility Operating License Nos. DPR–65 and NPF–49: These amendments revise the operating licenses and Millstone, Unit 2, Technical Specifications.

Date of initial notice in ***Federal Register:*** October 24, 2000 (65 FR 63630).

The October 12 and November 8, 2000, and February 16, 2001, supplements provided clarifying information and did not expand the scope of the application as originally published.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 5, 2001.

Brief description of amendment: The amendment revised the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs) 3.1.d.2 to reduce the maximum allowable leakage of primary system reactor coolant to the secondary system from 500 gallons per day (gpd) through any one steam generator to 150 gpd through any one steam generator. In addition, the amendment removes reference to the voltage based repair criteria.

Date of issuance: March 27, 2001.

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

Amendment No.: 153.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11062).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 13, 2000, as supplemented April 3, 2001.

Brief description of amendment: The amendment changes License Condition 2.C.4 to conform to NRC Generic Letter (GL) 86–10, ''Implementation of Fire Protection Requirements.'' The amendment also relocates the Fire Protection Program (FPP) elements from the Technical Specifications to the licensee-controlled FPP, in accordance with GL 86–10 and GL 88–12, ''Removal of Fire Protection Requirements from Technical Specifications.''

Date of issuance: April 5, 2001.

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

Amendment No.: 119.

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

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7684).

The April 3, 2001, letter was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: October 18, 2000.

Brief description of amendment: The amendment revised Section 2.1.2, Figures 2–1A and 2–1B, and the associated Bases of the Fort Calhoun Station Technical Specifications to extend the existing pressure-temperature (P–T) curves from 20 effective full power years (EFPY) to 24.25 EFPY. Additionally, the amendment deletes Figure 2–3, ''Predicted Radiation Induced NDTT Shift'' and updates the fluence analysis for projecting RT_{NDT} at 24.25 EFPY.

Date of issuance: March 27, 2001.

Effective date: March 27, 2001, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 197.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81925).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: April 14, 2000, as supplemented by letters dated June 2, July 28, and December 1, 2000, and January 31, 2001.

Brief description of amendment: The amendment changed the surveillance requirements for laboratory testing of the charcoal adsorbers for the control room, the spent fuel pool storage area and the safety injection pump rooms. In addition, the amendment deletes the laboratory testing requirements for the containment charcoal adsorbers. The changes comply with the guidance of Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: April 4, 2001.

Effective date: April 4, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 198.

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

Date of initial notice in ***Federal Register:*** March 5, 2001 (66 FR 13355).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 4, 2001.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: May 15, 2000, as supplemented on August 25, 2000.

Brief description of amendments: The amendments revise Technical Specifications (TSs) requirements to test the remaining diesel generators (DGs) when one of the two independent offsite power sources is inoperable, as described in Section 3/4.8.1, Action a; and when a DG is inoperable for other than preventive maintenance reasons, as described in Section 3/4.8.1, Action b, of the TSs. The amendments also expand the DG loading band from existing 2500–2600 KW to 2330–2600 KW for the monthly, 6-month, and the 2-hour loaded prerequisite for the hot restart tests, and correct an administrative oversight.

Date of issuance: April 2, 2001.

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

Amendment Nos.: 242 and 223.

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

Date of initial notice in ***Federal Register:*** July 12, 2000 (65 FR 43052).

The August 25, 2000, supplement did not change the conclusions made in the Commission's initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 11, 2001.

Brief description of amendments: The amendments delete Technical Specifications (TS) Section 5.5.2.2, ''Post Accident Sampling Program,'' for San Onofre Nuclear Generating Station, Units 2 and 3, and thereby eliminate the requirements to have and maintain the post-accident sampling systems (PASS). The amendments also revise TS 5.5.2.8, ''Primary Coolant Sources Outside Containment Program,'' to reflect the elimination of PASS. Additionally, the amendments delete PASS-related License Conditions 2.c(19)i for Unit 2 and 2.C.(17)d for Unit 3.

Date of issuance: March 26, 2001.

Effective date: March 26, 2001, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2—178; Unit 3–169.

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

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11063).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 5, 2001.

Brief description of amendment: The amendment authorizes a one cycle delay in removal of the second capsule.

Date of issuance: April 2, 2001.

Effective date: April 2, 2001.

Amendment No.: 271.

Facility Operating License No. DPR–52: The amendment revises the Reactor Vessel Material Surveillance Program.

Date of initial notice in ***Federal Register:*** February 28, 2001.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 22, 2001 (TS 00–01).

Brief description of amendments: These amendments revised the Technical Specifications (TSs) by revising the surveillance test requirement to assess flow blockage in the ice condenser containment.

Date of issuance: March 22, 2001.

Effective date: March 22, 2001.

Amendment Nos.: 267 and 258.

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

Date of initial notice in ***Federal Register:*** February 7, 2001 (66 FR 9388).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 22, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 26, 2000.

Brief description of amendment: This amendment revises Technical Specification (TS) requirements regarding secondary containment systems, including the Standby Gas Treatment System. The affected TS sections are 1.0, Definitions; 3/4.7.B, Standby Gas Treatment System; and 3/4.7.C, Secondary Containment System. In addition, a new TS section, 3/4.7.E, Reactor Building Automatic Ventilation System Isolation Valves is proposed. Some of the proposed changes are administrative in nature and do not affect the technical aspects of the requirements. Associated changes to the TS Bases are also being made to conform to the changed TS. The proposed changes provide certain additional flexibility in operations when equipment is made or found to be inoperable, while also ensuring appropriate actions are taken to place the plant in a safe condition under such conditions.

Date of Issuance: March 23, 2001.

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

Amendment No.: 197.

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

*Date of initial notice in **Federal Register:*** October 18, 2000 (65 FR 62394).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 23, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 25, 2000.

Brief description of amendment: The amendment revises the 125-volt DC station battery system Technical Specifications Section 3.10.A.2.b to reflect the availability of a second, fully qualified battery charger, for each main station battery system.

Date of Issuance: March 27, 2001.

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

Amendment No.: 198.

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

*Date of initial notice in **Federal Register:*** December 13, 2000 (65 FR 77928).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 7, 2000.

Brief description of amendment: The amendment revises Technical Specification 3.5.A.1 by adding a note regarding operability of the Low Pressure Coolant Injection system (LPCI) under certain restrictive conditions. The subject change would provide a clarification of system operability that would result in additional flexibility in operations during hot shutdown conditions.

Date of Issuance: March 30, 2001.

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

Amendment No.: 199.

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

*Date of initial notice in **Federal Register:***

January 24, 2001 (66 FR 7686).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 30, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 10th day of April 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–9320 Filed 4–17–01; 8:45 am]

BILLING CODE 7590–01–P

SMALL BUSINESS ADMINISTRATION

[Declaration of Disaster #3332]

Commonwealth of Massachusetts

As a result of the President's major disaster declaration on April 10, 2001, I find that Essex, Middlesex and Norfolk Counties constitute a disaster area due to damages caused by flooding and severe storms beginning March 5, 2001 and continuing. Applications for loans for physical damage as a result of this disaster may be filed until the close of business on June 9, 2001, and for loans for economic injury until the close of business on January 9, 2002 at the address listed below or other locally announced locations: U.S. Small Business Administration, Disaster Area 1 Office, 360 Rainbow Blvd. South 3rd Fl., Niagara Falls, NY 14303.

In addition, applications for economic injury loans from small businesses located in the following contiguous counties in Massachusetts may be filed until the specified date at the above location: Bristol, Plymouth, Suffolk and Worcester; and Hillsborough and Rockingham counties in the State of New Hampshire; and Providence county in the State of Rhode Island.

The interest rates are:

	Percent
For Physical Damage:	
Homeowners with credit available elsewhere	7.000
Homeowners without credit available elsewhere	3.500
Businesses with credit available elsewhere	8.000
Businesses and non-profit organizations without credit available elsewhere	4.000
Others (including non-profit organizations) with credit available elsewhere	7.000
For Economic Injury:	
Businesses and small agricultural cooperatives without credit available elsewhere	4.000

The number assigned to this disaster for physical damage is 333206 and for economic injury the number is 9L4300. The number assigned for economic injury for New Hampshire is 9L4400 and for Rhode Island is 9L4500.

(Catalog of Federal Domestic Assistance Program Nos. 59002 and 59008.)

Dated: April 11, 2001.

Herbert L. Mitchell,

Associate Administrator for Disaster Assistance.

[FR Doc. 01–9599 Filed 4–17–01; 8:45 am]

BILLING CODE 8025–01–P

SMALL BUSINESS ADMINISTRATION

[License No.: 05/05–0228]

Mezzanine Capital Partners, Inc.; Notice of Surrender of License

Notice is hereby given that *Mezzanine Capital Partners, Inc., located at 150 South 5th Street, Suite 1720, Minneapolis, MN 55402*, has surrendered its license to operate as a small business investment company under the Small Business Investment Act of 1958, as amended (the Act). *Mezzanine Capital Partners, Inc.* was