longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@hrc.gov or dkw@nrc.gov.

Dated: November 24, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–30465 Filed 11–27–00; 10:45 am]

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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 November 6, 2000, through November 16, 2000. The last biweekly notice was published on November 15, 2000.

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC's 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 December 29, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, 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: Docketing and Services 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of amendment request: October 6, 2000 (U–603329).

Description of amendment request: The proposed amendment would relocate Technical Specification Figure 3.6.4.1–1, ''Secondary Containment Drawdown Time for 1500 cfm Boundary Leakage'' to plant procedures.

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

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

The proposed Technical specification (TS) change eliminates an inconsistency between the Secondary Containment surveillance requirement (SR 3.6.4.1.4) and the associated Bases. The proposed change (1) revises the wording of SR 3.6.4.1.4 to verify the time to draw down the secondary containment to ≥0.25 inch water gauge for each standby gas treatment (SGT) subsystem is within the required time; and (2) relocates the specific acceptance criteria (existing TS Figure 3.6.4.1–1) to plant procedures and the TS Bases.

The scope of the proposed change is thus limited to the affected SR. No changes to plant equipment or the plant design are involved. The affected SR, as are surveillances in general, is not an initiator to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased.

The proposed change impacts SR 3.6.4.1.4 but does not change its intent or the associated acceptance criteria. Thus, the components and structural integrity being tested will still be required to be maintained Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification (TS) change eliminates an inconsistency between the Secondary Containment surveillance requirement (SR 3.6.4.1.4) and the associated Bases. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. No new failure modes are thus introduced by the proposed change. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification (TS) change eliminates an inconsistency between the Secondary Containment Integrity surveillance requirement (SR 3.6.4.1.4) and the associated Bases. The revised wording of the Surveillance Requirement and the relocation of the acceptance criteria to plant procedures and TS Bases have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. The relocation of the acceptance criteria is consistent with the Bases previously approved in Amendment 21. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, this 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: Anthony J. Mendiola.

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

Date of amendment request: October 6, 2000 (U–603332).

Description of amendment request: The proposed amendment would remove from the Technical Specification surveillance requirements the minimum operating time specified for the containment/hydrogen mixing system.

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

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

The proposed Technical Specification (TS) change deletes the minimum run time requirement (of 15 minutes) for the Containment/Drywell Hydrogen Mixing System as denoted in Surveillance Requirement 3.6.3.3.1. Satisfying a TS Surveillance Requirement ensures that the associated system will function to mitigate the consequences of an accident, and, as such is not an initiator of an accident or malfunction. Therefore, since such a test requirement or operation is not an initiator to any accident previously evaluated, the probability of an accident previously evaluated is not significantly increased.

In addition, the equipment being tested is still required to be operable and capable of performing its accident mitigation function assumed in the accident analysis. Eliminating the time requirement from the Surveillance Requirement does not reduce or relax the requirements to ensure that all controls are functioning properly. Also, it does not reduce or relax the requirements for ensuring the degraded conditions such as piping blockage, compressor failure or excessive vibration can be detected for corrective action. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed TS change, as denoted in Surveillance Requirement 3.6.3.3.1, does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. No new potential accident initiators are therefore introduced by the proposed change. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS change to delete the minimum run time requirement in Surveillance Requirement 3.6.3.3.1 does not result in a significant reduction in the margin of safety. As provided in the justification for the proposed change, the 15-minute run time acceptance criterion is not necessary to ensure that the Containment/Drywell Hydrogen Mixing can perform its required function. Thus, if a margin of safety can be ascribed to proper operation of this system for LOCA mitigation, the system will continue to be tested in a manner that gives confidence that it can perform its assumed safety function. Therefore, this 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: Anthony J. Mendiola.

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

Date of amendment request: August 29, 2000.

Description of amendment request: The proposed amendment would revise the Technical Specifications in the ''Administrative Controls'' section to certain position titles and the Shift Technical Advisor (STA) staffing requirement to allow one of the required on-shift Senior Reactor Operator (SRO) positions to be combined with the required STA position so as to serve in a dual role SRO/STA position.

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

The [proposed] changes do not affect the purpose, function, performance, operability, inspection or testing of and does not make any physical or procedural changes to plant systems, structures or components. Also, all existing technical specification limiting conditions for operation and surveillance requirements are retained.

TSCR [Technical Specifications Change Request] makes administrative changes to certain position titles without changing the technical requirements for position responsibilities.

TSCR also changes the Shift Technical Advisor (STA) staffing requirement to allow one of the required on-shift Senior Reactor Operator (SRO) positions to be combined with the required STA position so as to serve in a dual-role SRO/STA position as encouraged by the NRC in Option 1 of Generic Letter 86–04, ''Policy Statement on Engineering Expertise On Shift'', dated February 13, 1986.

Implementation of the proposed dual-role SRO/STA change will result in personnel with enchanced operational knowledge being assigned to perform the STA function of providing accident assessment expertise, and analyzing and responding to off normal occurrences when needed. The NRC's stated preference in the October 28, 1985, ''Policy Statement on Engineering Expertise on Shift'', indicates that the NRC has concluded that the individual filling the dual-role SRO/STA position may perform these functions better than a non-licensed individual filling the STA position even when the SRO/STA is concurrently functioning as one of the required shift SROs.

Therefore, since no physical or procedural changes are being made to existing plant systems, structures or components and since the position title changes are administrative in nature and the function and responsibilities of the STA will be executed by an appropriately qualified individual filling the dual-role SRO/STA position, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The [proposed] changes do not affect the purpose, function, performance, operability, inspection or testing of and does not make any physical or procedural changes to plant systems, structures or components. Also, all existing technical specification limiting conditions for operation and surveillance requirements are retained.

TSCR 277 makes administrative changes to certain position titles without changing the technical requirements for the position responsibilities.

TSCR 277 also changes the Shift Technical Advisor (STA) staffing requirement to allow one of the required on-shift Senior Reactor Operator (SRO) positions to be combined with the required STA position so as to serve in a dual-role SRO/STA position as encouraged by the NRC in Option 1 of Generic Letter 86–04, ''Policy Statement on Engineering Expertise On Shift'', dated February 13, 1986.

Therefore, since no physical or procedural changes are being made to existing plant systems, structures or components and since the position title changes are administrative in nature and the function and responsibilities of the STA will be executed by an appropriately qualified individual filling the dual-role SRO/STA position, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The [proposed] changes do not affect the purpose, function, performance, operability, inspection or testing of and does not make any physical or procedural changes to plant systems, structures or components. Also, all existing technical specification limiting conditions for operation and surveillance requirements are retained.

TSCR 277 makes administrative changes to certain position titles without changing the technical requirements for the position responsibilities.

TSCR 277 also changes the Shift Technical Advisor (STA) staffing requirement to allow one of the required on-shift Senior Reactor Operator (SRO) positions to be combined with the required STA position so as to serve in a dual-role SRO/STA position as encouraged by the NRC in Option 1 of Generic Letter 86–04, ''Policy Statement on Engineering Expertise On Shift'', dated February 13, 1986.

Therefore, since no physical or procedural changes are being made to existing plant systems, structures or components and since the position title changes are administrative in nature and the function and responsibilities of the STA will be executed by an appropriately qualified individual filling the dual-role SRO/STA position and shift staffing required by TS 6.2.2.2 and 10 CFR 50.54(m)(2) will [continue] to be maintained, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in [a] margin of safety.

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

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

NRC Section Chief: M. Gamberoni.

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

Date of amendment request: October 5, 2000.

Brief description of amendment: The proposed changes would revise Technical Specifications (TSs) Sections 3.2.1.a, 3.2.1.e, and 3.2.1.f to relocate administrative controls to the Quality Assurance Program Description (QAPD).

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

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

No. The proposed changes are administrative in nature. The changes involve Section 3.2.1.a, referencing the QAPD instead of the IP#2 [Indian Point Nuclear Generating Station, Unit 2] UFSAR [Updated Final Safety Analysis Report], deleting Section 3.2.1.e and having Section 3.2.1.f, refer to the QAPD for the administrative controls. These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

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

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

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

No. The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or the physical design, the IP#1 [Indian Point Nuclear Generating Station, Unit 1] FSAR [Final Safety Analysis Report] or the IP#2 UFSAR design basis, accident assumptions, or IP#1 Technical Specification Bases are not affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for Licensee: Brent L. Brandenberg, Esq., Assistant General Counsel, Consolidated Edison Company of New York, Inc., 4 Irving Place-1830, New York, NY 10003.

NRC Section Chief: Michael T. Masnik.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: October 12, 2000.

Description of amendment request: The amendment changes the facility name of WNP–2 to Columbia Generating Station in all the applicable portions of the Operating License including Appendix A (Technical Specifications) and Appendix B (Environmental Protection Plan). In addition, the proposed change makes editorial changes to Technical Specification Figure 4.1–1, Site Area Boundary.

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

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

No. This request involves an administrative change only. The Operating License (OL) and Technical Specification Figure 4.1–1, Site Area Boundary, are being changed to reflect the new name of the facility. In addition, editorial changes are being made to Figure 4.1–1 for clarification. No actual plant equipment or accident analyses are affected by the proposed change. Therefore, this request will have no impact on the probability or consequence of any type of accident previously evaluated.

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

No. This request involves an administrative change only. The OL and Technical Specification Figure 4.1–1 are being changed to reflect the new name of the facility. In addition, editorial changes are being made to Figure 4.1–1 for clarification. No actual plant equipment or accident analyses are affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, this request will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

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

No. Margin of safety is associated with confidence in the ability of the fission product barriers (*i.e.,* fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves an administrative change only. The OL and Technical Specification Figure 4.1–1 are being changed to reflect the new name of the facility. In addition, editorial changes are being made to Figure 4.1–1 for clarification.

No actual plant equipment or accident analyses are affected by the proposed change. Additionally, the proposed change will not relax any criteria used to establish safety limits, will not relax any safety system settings, or will not relax the bases for any limiting conditions of operation. Therefore, this proposed change will not impact margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: October 30, 2000.

Description of amendment request: Surveillance Requirement (SR) 3.6.1.3.8 currently requires verification of the actuation capability of each excess flow check valve (EFCV) every 24 months. The proposed change is to relax the SR frequency by allowing a ''representative sample'' of reactor instrument line EFCVs to be tested every 24 months, such that each reactor instrument line EFCV will be tested at least once every 10 years (nominal). The proposed change will also result in limiting the SR to only the reactor instrument line EFCVs.

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

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

The current SR frequency requires each reactor instrument line EFCV to be tested every 24-months. The reactor instrument line EFCVs at WNP–2 are designed so that they will not close accidentally during normal operation, but will close if a rupture of the instrument line is indicated downstream of the valve, and have their status indicated in the control room. This proposed change allows a reduced number of reactor instrument line EFCVs to be tested every 24-months. There are no physical plant modifications associated with this change. Industry operating experience demonstrates a high reliability of these valves. Neither reactor instrument line EFCVs nor their failures are capable of initiating previously evaluated accidents; therefore, there can be no increase in the probability of occurrence of an accident regarding this proposed change.

Reactor instrument lines connecting to the reactor coolant pressure boundary are equipped with EFCVs and also have a flow-restricting orifice inside containment and upstream of the EFCV. The consequences of an unisolable rupture of such an instrument line has been previously evaluated in WNP–2 FSAR [Final Safety Analysis Report] 15.6.2. The instrument lines that penetrate primary containment conform to Regulatory Guide 1.11 (WNP–2 FSAR 7.1.2.4). Those instrument lines are Seismic Category I and terminate in instruments that are Seismic Category I (reference WNP–2 FSAR Table 6.2–16 note 27).

The sequence of events in WNP–2 FSAR Section 15.6.2.2 for a reactor instrument line break assumes a continuous discharge of reactor water through the instrument line until the reactor vessel is cooled and depressurized (5 hours). Although not expected to occur as a result of this change, the postulated failure of an EFCV to isolate as a result of reduced testing is bounded by this previous evaluation. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the consequences of an accident previously evaluated as a result of this change.

The containment atmosphere and suppression pool instrument line EFCVs are required to remain open to sense containment atmosphere and suppression pool level conditions during postulated accidents. They are not required to close during an instrument line break assumed during normal plant operation nor is their design capable of closing during normal plant conditions. These EFCVs do not meet the criteria for inclusion in 10 CFR 50.36(c)(3) as they have no active safety function and thus relocation of their testing requirements to the FSAR cannot effect the probability of an increase in the consequences of an accident previously evaluated.

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

This proposed change allows a reduced number of reactor instrument line EFCVs to be tested each operating cycle and that the testing requirements for containment atmosphere and suppression pool instrument line EFCVs be relocated to the FSAR. No other changes in requirements are being proposed. Industry operating experience demonstrates the high reliability of these valves. The potential failure of a reactor instrument line EFCV to isolate by the proposed change in testing is bounded by the previous evaluation of an instrument line rupture. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

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

The consequences of an unisolable rupture of an instrument line has been evaluated in WNP–2 FSAR Section 15.6.2 in accordance with the requirements of Regulatory Guide 1.11. That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (5 hours). The only margin of safety applicable to this proposed change is considered to be that implied by this evaluation. Since a continuous discharge was assumed in this evaluation, any potential failure of a reactor instrument line EFCV to isolate as a result of reduced testing frequency is bounded by existing analysis and does not involve a significant reduction in the margin of safety.

There is no accident for which the containment atmosphere or suppression pool instrument line EFCVs are designed to actuate to the isolation position for mitigation. A postulated break of a containment atmosphere or suppression pool instrument line under normal operating conditions would not result in a condition that would create the ability for these EFCVs to operate because neither the containment pressure nor the suppression pool level head would be sufficient to result in their actuation. As these EFCVs have no active design or safety function, the relocation of testing requirements would not involve a significant reduction in the margin of safety. A postulated break of any instrument line simultaneously with a loss of coolant accident is beyond the design basis for 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 C. Poindexter, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: February 2, 2000, as supplemented on August 11 and September 18, 2000.

Description of amendment request: The proposed amendment would approve the license termination plan for the SNEF.

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

The proposed change is necessary to achieve the decommissioning objective of terminating the license and releasing the site for unrestricted use. As such, the proposed change:

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated since accidents which might occur during the active decommissioning phase of the SNEC facility are bounded by the twelve accidents addressed in section 3.0 of the Updated Safety Analysis Report (USAR). The accident analysis addressed in the USAR demonstrate that no adverse public health and safety impacts are expected from accidents that

might occur during decommissioning operations at the SNEC facility. The greater part of radioactively contaminated materials and components originally located in the SNEC facility Containment Vessel are no longer on site, having been shipped as radioactive waste.

Implementation of the SNEC License Termination Plan involves a continuation of the decommissioning process including the final status survey activity to be performed prior to site closeout at the end of the dismantlement phase. These activities do not involve a significant increase in either the probability or consequences of an accident previously evaluated.

2. Will not create the possibility of a new or different kind of accident from any accident previously evaluated. Accidents previously evaluated in the USAR access different methods of dispersing radioactive material to the environment, which include a loss of support systems and external events. Remaining dismantlement activities and final status survey work described in the License Termination Plan are similar to those previously performed and will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will not involve a significant reduction in a margin of safety. The Technical Specifications currently in place at the SNEC facility were developed to safely decommission the SNEC facility. Issuance of the proposed amendment would not reduce the controls established by the technical specifications for activities performed at the SNEC facility. The proposed License Amendment establishes additional controls to ensure License Termination Plan activities are performed effectively. Thus, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensees 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 the Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, N.W., Washington, DC 20037.

NRC Branch Chief: Ledyard B. Marsh.

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

Date of amendment request: June 1, 2000.

Description of amendment request: The licensee is proposing to relocate Technical Specifications 3.3.3.2, ''Instrumentation, Movable Incore Detectors''; 3.3.3.3, ''Instrumentation, Seismic Instrumentation''; 3.3.3.4, ''Instrumentation, Meteorological Instrumentation''; 3.3.3.8, ''Loose-Part Detection System''; and 3.3.4, ''Turbine Overspeed Protection'' and Index Pages vi and vii to the Technical Requirements Manual (TRM). The Bases of the affected Technical Specifications will be modified to address the proposed changes.

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

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

The proposed changes to relocate the movable incore detector instrumentation, seismic monitoring instrumentation, meteorological monitoring instrumentation, loose-part detection instrumentation, and turbine overspeed protection instrumentation from the Technical Specifications to the TRM 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] DBAs will not change. In addition, the movable incore detector instrumentation, seismic monitoring instrumentation, meteorological monitoring instrumentation, and loose-part detection instrumentation are not accident initiators and cannot cause an accident. For the turbine overspeed protection instrumentation, the DBAs and transients include a variety of system failures and conditions which might result from turbine overspeed events and potential missiles striking various plant systems and equipment. However, in view of the low likelihood of the generation of turbine missiles, the turbine overspeed protection instrumentation does not serve a primary protective function. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to relocate the movable incore detector instrumentation, seismic monitoring instrumentation, meteorological monitoring instrumentation, loose-part detection instrumentation, and turbine overspeed protection instrumentation do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any [structure, system, or component] SSC functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification and Bases changes will relocate the requirements for the movable incore detector instrumentation, seismic monitoring instrumentation, meteorological monitoring instrumentation, loose-part detection instrumentation, and turbine overspeed protection instrumentation from Technical Specifications to the TRM. Any future changes to the relocated requirements will be in accordance with 10 CFR 50.59 and approved station procedures. 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 DBAs will not change. In addition, the relocated requirements do not meet any of the 10 CFR 50.36c(2)(ii) criteria on items for which Technical Specifications must be established. Therefore, there will be no 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 29, 2000 as supplemented on October 16, 2000.

Description of amendment request: The proposed changes would modify Technical Specification (TS) Sections 3.3.2, ''Instrumentation—Engineered Safety Features Actuation System Instrumentation;'' 3.7.7, ''Plant Systems—Control Room Emergency Ventilation System;'' 3.7.8, ''Plant Systems—Control Room Envelope Pressurization System;'' 3.7.9, ''Plant Systems—Auxiliary Building Filter System;'' 3.9.1.1, ''Refueling Operations—Boron Concentration,'' 3.9.1.2; ''Refueling Operations—Boron Concentration;'' 3.9.2, ''Refueling Operations—Instrumentation;'' 3.9.4, ''Refueling Operations—Containment Building Penetrations;'' 3.9.9, ''Refueling Operations—Containment Purge and Exhaust Isolation System;'' 3.9.10, ''Refueling Operations—Water Level—Reactor Vessel;'' and 3.9.12, ''Refueling Operations—Fuel Building Exhaust Filter System.'' Some of these proposed changes are associated with the revised fuel handling accident analysis, and integrity of the Control Room and the Fuel Building boundaries. Several administrative changes are also proposed to reflect Millstone Unit 3 terminology, removal of unnecessary information and to eliminate confusion by providing consistency between limiting conditions for operation, action requirements, and Surveillance Requirements. The proposed Technical Specifications changes associated with the revised containment fuel handling accident analysis results in an increase in the consequences of a containment fuel handling accident since the current analysis of a containment fuel handling accident does not assume the release of any radioactive material from containment. The revised analysis assumes a release of radioactive material because it assumes both personnel access hatch doors are open and at least one hatch door is closed within 10 minutes of a fuel handling accident inside containment.

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

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

Technical Specification Changes Associated with Analyses Changes

The proposed Technical Specification changes associated with the revised fuel handling accident analyses will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability of an accident previously evaluated.

The proposed Technical Specification changes associated with the revised fuel handling accidents analyses will increase the associated consequences. The increased consequences are the result of a revised plant configuration and revised calculation assumptions, not the result of the addition of any new plant equipment. The current Fuel Handling Accident Inside Containment (FHAIC) analysis assumes the containment is isolated, or will be isolated, prior to any release. The revised FHAIC analysis will allow both containment personnel access hatch doors to remain open, under administrative control, during core alterations and irradiated fuel movement inside containment. This may result in a radioactive release if a fuel handling accident were to occur. The revised FHAIC analysis demonstrates that the magnitude of the potential release is small and bounded by the consequences of the Design Basis Loss of Coolant Accident. The increase in the consequences of the revised Fuel Handling Accident Inside the Spent Fuel Pool (FHAISFP) analysis due to the revised calculation assumptions is small. Therefore, the proposed changes will not result in a significant increase in the consequences of an accident previously evaluated.

Other Technical Specification Changes

The proposed Technical Specification changes not associated with the revised fuel handling accidents analyses affect the limiting conditions for operation (LCOs), applicability, action requirements, and surveillance requirements of numerous specifications associated with plant operating restrictions, accident mitigation functions, and accident mitigation equipment. The affected operating restrictions, accident mitigation functions, and accident mitigation equipment are not accident initiators. The proposed changes will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability of an accident previously evaluated.

The proposed LCO and applicability changes are consistent with the design basis accident analyses, including the revised fuel handling accident analyses. (The proposed change to the LCO for containment penetrations, which will allow both personnel access hatch doors to remain open during core alterations and irradiated fuel movement inside containment will result in an increase in the consequences of a FHAIC as previously discussed.) This will ensure that the accident mitigation functions and associated equipment are available for accident mitigation as assumed in the associated analyses. As a result, the accident analysis assumptions and mitigation methods will not be adversely affected by these changes. Therefore, the proposed changes will not result in a significant increase in the consequences of an accident previously evaluated.

The additional proposed changes to the Technical Specifications that will standardize terminology, relocate information to the Bases, remove extraneous information, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Technical Specifications do not impact any system or component that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any 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 response of the plant and the operators following an accident will not be significantly different. In addition, the proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Title 10 of the Code of Federal Regulations Part 100 establishes the accident exposure limits (300 rem thyroid and 25 rem whole body) for the Exclusion Area Boundary and Low Population Zone. The radiological consequences resulting from the Technical Specification changes associated with the revised fuel handling accident analyses are well within these limits. The radiological consequences to the Control Room Operators resulting from the Technical Specification changes associated with the revised fuel handling accident analyses are also within the GDC 19 limit. Since these limits will not be exceeded and these limits establish the margin of safety in the plant's current licensing basis, the proposed changes will not result in a significant reduction in a margin of safety.

The proposed Technical Specification LCO, applicability, action requirement, and surveillance requirement changes not associated with the revised fuel handling accidents analyses do not adversely affect equipment design or operation. In addition, the proposed allowed outage times and shutdown times are consistent with times already contained in the Millstone Unit No. 3 Technical Specifications. Therefore, these changes will not result in a significant reduction in a margin of safety.

The additional proposed changes to the Technical Specifications that will standardize terminology, relocate information to the Bases, remove extraneous information, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional changes will not result in a significant reduction in a margin of safety.

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

Attorney for licensee: 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 No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: October 12, 2000.

Description of amendment request: The proposed amendment would make changes to Hope Creek Generating Station (HCGS) Technical Specifications (TS) and Bases associated with the drywell vacuum breakers and the suppression pool vacuum breakers. The proposed changes are intended to provide consistency between the HCGS TS and the Standard Technical

Specifications (STS) (NUREG–1433). These changes include revising or deleting specific Limiting Conditions for Operation and Surveillance Requirements and include relocating information from these sections to the Bases. In addition, a change to the Containment Systems Surveillance Requirements was proposed to correct the hierarchical format of that section.

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

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

The function of the suppression-chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell by allowing air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Steam condensing in the drywell as a result of a primary system rupture results in the most severe pressure transient.

The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when the primary containment depressurizes below the reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression-chamber-to-drywell vacuum breakers. A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere.

Each proposed change to the vacuum breaker TS was categorized by the licensee as either administrative, more restrictive, less restrictive, or as a relocation. In addition, the licensee made changes to the bases to capture the information removed from the associated Action Steps and to be consistent with the STS. The administrative changes eliminate, replace, or add words or phrases, to provide clarity or to achieve consistency with the STS. The more restrictive changes reduce the number of vacuum breakers allowed to be open or reduces the amount of time allowed to close the open valves. The less restrictive changes: (1) Eliminate the surveillance requirements associated with the vacuum breaker position indicators; (2) reduce the frequency of vacuum breaker position verification; (3) increase the time requirement for functional testing subsequent to steam discharged to the suppression chamber from the safety-relief valves; (4) increase the number of allowable inoperable valves in one vacuum breaker assembly; (5) eliminate repetitious visual inspections; and (6) eliminate channel calibration as a means to determine operability of the inboard isolation valve auto-open control system. The relocation changes move information from the action steps to the bases.

The licensee stated in their October 12, 2000, application that neither the vacuum breakers, the vacuum breaker position indication, nor the vacuum breaker actuation system are initiators of any analyzed event. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated. In addition, the licensee's application states that the proposed changes will ensure the operability of the vacuum breakers, will provide assurance that the containment integrity and venting capability are maintained or restored within 1 hour, and that sufficient vacuum breakers will remain operable to mitigate the assumed accidents. Therefore, there is no significant increase in the consequences of an accident previously evaluated. The licensee further stated that any future changes to the licensee-controlled documents containing relocated requirements will be evaluated in accordance with the PSEG Nuclear 10 CFR 50.59 program. Consequently, no significant increase in the consequences of an accident previously evaluated will be allowed without prior Nuclear Regulatory Commission (NRC) approval. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The licensee stated in their application that the proposed change does not involve a physical alteration of the plant and does not introduce any new modes of plant operation. In addition, the licensee stated that any resulting changes to the operation of the plant will be consistent with assumptions made in the safety analysis. Therefore the proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The licensee stated in their application that the proposed change does not impact any safety analysis assumptions and will provide assurance that the containment integrity and venting capability are maintained or restored within 1 hour. The licensee further stated that Hope Creek experience has shown that the change in surveillance frequency of vacuum breaker position is not a significant change in operating practice. In addition, the operability of the vacuum breakers is not adversely affected by steam discharged through the safety relief valves (SRVs) and does not pose an immediate operability concern. Consequently, the potential impact from the proposed increase in the amount of time during which to perform functional testing subsequent to an SRV lift is minimal. Therefore, there is no significant reduction in a margin of safety. The licensee further stated that any future changes to the licensee-controlled documents containing relocated requirements will be evaluated in accordance with the PSEG Nuclear 10 CFR 50.59 program. Consequently, no significant reduction in a margin of safety will be allowed without prior NRC approval. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 30, 2000 (TS–407).

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to remove the term "maximum pathway" from the main steam isolation valve (MSIV) leakage rate Surveillance Requirement (SR) 3.6.1.3.10. This proposed change would provide consistency with 10 CFR Part 50 Appendix J leak rate testing terminology for evaluating MSIV leakage rates.

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

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

The proposed change to eliminate the words "maximum pathway" does not affect any plant system or component, and does not impact operator performance or procedures. The leak rate testing of the MSIVs will continue to be performed in accordance with 10 CFR 50 Appendix J in a manner consistent with the guidance on leak rate testing presented in industry guidance documents and in the Standard TS. The change does not impact the design basis accident analyses presented in the Final Safety Analysis Report (FSAR). This proposed TS change is considered administrative in that no changes in leak testing methods or in disposition of leak rate results are involved. Therefore, the proposed 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.

No changes in accident analysis are involved, so the consequences of accidents will remain within the accident analysis described in the FSAR. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not affect any plant system or component, and does not have any impact on plant operation. No changes in accident analyses are involved, therefore, the proposed change does not involve a significant reduction in the margin of safety as currently defined in the bases of the applicable TS section or in the FSAR. For these reasons, the proposed amendment 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET I0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: November 6, 2000 (TS–411).

Description of amendment request: The proposed amendment would revise the technical specifications to allow two Residual Heat Removal (RHR) suppression pool cooling subsystems to be inoperable for 8 hours.

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

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

The change does not result in any hardware or operating procedure changes. The RHR Suppression Pool Cooling subsystems are not assumed to be initiators of any analyzed event. This change allows an additional 8 hours to restore required RHR Suppression Pool Cooling subsystem(s) prior to requiring the initiation of a unit shutdown. The proposed 8 hour Completion Time provides some time to restore required subsystem(s) to Operable status, yet is short enough that operating an additional 8 hours is not a significant risk.

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

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

The possibility of a new or different kind of accident from any previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve a physical modification to the plant.

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

The increased time allowed for restoring required inoperable RHR Suppression Pool Cooling subsystems is acceptable based on the small probability of an event requiring the inoperable suppression pool cooling subsystems to function and the desire to restore required subsystems prior to requiring the initiation of a plant shutdown. Delaying a plant shutdown will minimize the potential for a scram which then could result in a need for a subsystem when it is inoperable. As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing additional time to restore required subsystem(s), thus avoiding potential plant transients during shutdown. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET I0H, Knoxville, Tennessee 37902

NRC Section Chief: Richard P. Correia.

Three Mile Island Nuclear Station, Unit 2, Docket No. 50–320, Middletown, Pennsylvania

Date of amendment request: August 9, 2000.

Brief description of amendment request: The proposed technical specifications change request (TSCR) is to revise Three Mile Island Nuclear Generating Station, Unit 2 (TMI–2), Technical Specifications Sections 6.5.3.2, 6.5.4.1, 6.5.4.2.a, 6.5.4.2.b, 6.5.4.3, 6.5.4.3.c, 6.5.4.4 and 6.5.4.6, to eliminate the reference to Independent Onsite Safety Review Group (IOSRG) and to define the performance of the IOSRG function by the nuclear quality assurance organization. Also, two titles that no longer exist (Manager, TMI–2 Department and division vice president) were corrected. These administrative changes are similar to changes that have been already approved at other plants in Region I.

Basis for proposed no significant hazards consideration: As required by 10 CFR 50.91(a), the license 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 of occurrence or the consequences of an accident previously evaluated. The proposed changes do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR [Updated Final Safety Analysis Report] transient analyses. No Technical Specification Limiting Condition for Operation, Action Statement, or Surveillance Requirement is affected by any of the proposed changes. The proposed changes do not alter the design, function, or operation of any plant component. Therefore, the proposed amendment does not affect the probability of occurrence 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 previously evaluated. The proposed changes do not affect assumptions contained in plant safety analyses, the physical design and/or modes of plant operation defined in the plant operating license, or Technical Specifications that preserve safety analysis assumptions. The proposed changes do not introduce a new mode of plant operation or surveillance requirement, nor involve a physical modification to the plant. The proposed changes do not alter the design, function, or operation of any plant components. Therefore, the proposed amendment does not affect 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. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR transient analyses. No Technical Specification Limiting Condition for Operation, Action Statement, or Surveillance Requirement is affected. Therefore, the proposed amendment does not reduce the margin of safety.

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

Attorney For Licensee: Ernest L. Blake, Jr. Esq., Shaw, Pittman, Potts & Trowbridge 2300 N. Street, N.W., Washington, DC 20037.

NRC Section Chief: Mike Masnik.

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

Date of amendment request: October 25, 2000.

Description of amendment request: The proposed amendment would make editorial and administrative changes to the Technical Specifications (TS). These changes correct spelling and grammatical errors, correct references, eliminate excessive detail related to specifying a job title, revise position titles, consolidate pages and generalize statements allowing Nuclear Regulatory Commission (NRC) approved alternatives to specified requirements.

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

1. The operation of the Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative or editorial in nature and do not involve any physical changes to the plant. The changes do not revise the methods of plant operation which could increase the probability or consequences of accidents. No new modes of operation are introduced by the proposed changes such that a previously evaluated accident is more likely to occur or more adverse consequences would result.

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

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

These changes are administrative or editorial in nature and do not affect the operation of any systems or equipment, nor do they involve any potential initiating events that would create any new or different kind of accident. There are no changes to the design assumptions, conditions, configuration of the facility, or manner in which the plant is operated and maintained.

The changes do not affect assumptions contained in plant safety analyses or the physical design and/or modes of plant operation. Consequently, no new failure mode is introduced.

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

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

There are no changes being made to the TS safety limits or safety system settings. The operating limits and functional capabilities of systems, structures and components are unchanged as a result of these administrative and editorial changes. These changes do not affect any equipment involved in potential initiating events or plant response to accidents. There is no change to the basis for any Technical Specification that is related to the establishment or maintenance of, a nuclear safety margin.

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

NRC Section Chief: James W. Clifford.

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

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

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

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

Date of application for amendments: October 30, 2000.

Brief description of amendments: Modify the Technical Specifications to allow a one-time-only increase in the diesel generator Action Completion Time from 72 hours to 10 days to facilitate potential repairs to an emergency diesel generator to improve reliability.

Date of publication of individual notice in the ***Federal Register:*** November 3, 2000 (65 FR 66266).

Expiration date of individual notice: December 4, 2000.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: September 20, 2000.

Brief description of amendment: The amendment allows placing a static VAR compensator into service with just one of the two protective subsystems operable.

Date of issuance: November 13, 2000.

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

Amendment No.: 136.

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

Date of initial notice in **Federal Register:** October 2, 2000 (65 FR 58829).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 21, as supplemented on June 14, September 26, and October 16, 2000.

Brief description of amendment: The proposed amendment revised the Technical Specifications to delete the reporting requirements for the core spray sparger inspection.

Date of Issuance: November 2, 2000.

Effective date: November 2, 2000 and shall be implemented within 30 days of issuance.

Amendment No.: 217.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications and License.

Date of initial notice in **Federal Register:** September 22, 2000 (65 FR 57404).

The June 14, September 26, and October 16, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 30, as supplemented on September 26, 2000.

Brief description of amendment: The proposed amendment revised the Technical Specifications (TSs) to establish that the existing Safety Limit Minimum Critical Power Ratio contained in TS 2.1.A is applicable for the next operating cycle (Cycle 18).

Date of Issuance: November 3, 2000.

Effective date: November 3, 2000 and shall be implemented within 30 days of issuance.

Amendment No.: 218.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 22, 2000 (65 FR 57406).

The September 26, 2000, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 3, 2000.

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: November 19, 1999, as supplemented July 18, 2000.

Brief description of amendments: The amendments revise Technical Specification 5.5.11.c, ''Ventilation Filter Testing Program (VFTP),'' to include the requirement for laboratory testing of Engineered Safety Feature Ventilation System charcoal samples per American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' in response to Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999.

Date of issuance: November 8, 2000.

Effective date: November 8, 2000, and shall be implemented within 45 days of the date of issuance.

Amendment Nos.: Unit 1–130, Unit 2–130, Unit 3–130.

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

Date of initial notice in **Federal Register:** March 8, 2000 (65 FR 12287).

The July 18, 2000, supplement provided clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 29, 2000, as supplemented by letters dated June 26 and August 18, 2000.

Brief description of amendments: The amendments revised the pressure-temperature (P–T) limits for heatup, cooldown, critical operation and inservice leak and hydrostatic test limitations for the reactor pressure vessel (RPV). The amendments replaced the current RPV P–T limit curves with three recalculated curves that are applicable to 32 effective full power years. The staff has approved the revised limits for an interim period not to exceed December 15, 2002.

Date of issuance: November 8, 2000.

Effective date: Immediately until December 15, 2002, to be implemented within 30 days.

Amendment Nos.: 144 and 130.

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

Date of initial notice in **Federal Register:** April 5, 2000 (65 FR 17911). The June 26 and August 18, 2000, submittals provided additional information that did not change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 28, 2000.

Brief description of amendments: The amendments revise License Condition 2.C.(37) for Unit 1 and License Condition 2.C.(21) for Unit 2, to specify the types of fuel movements that cannot be performed during refueling unless all control rods are fully inserted.

Date of issuance: November 9, 2000.

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

Amendment Nos.: 145 and 131.

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

*Date of initial notice in **Federal Register:*** June 14, 2000 (65 FR 37422).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 9, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 3, 1999, as supplemented by letters dated April 4, June 9, June 29, August 2, and August 16, 2000.

Brief description of amendment: The amendment authorized revision of the Safety Analysis Report (SAR) to increase the containment structural design pressure from 54 psig to 59 psig, revised Technical Specification (TS) Table 3.3–3 to add a containment spray actuation signal on high-high containment building pressure to terminate main feedwater and main steam flow from the unaffected steam generator, revised TS 3.6.1.4 and Figure 3.6–1 to change the allowable containment initial conditions to be consistent with analysis assumptions, and revised TS 6.15 to increase the calculated peak accident pressure in the containment leakage rate testing program from 54 psig to 58 psig. Related changes to the Bases were also made.

Date of issuance: November 13, 2000.

Effective date: As of the date of issuance to be implemented prior to the commencement of heatup from refueling outage 2R14.

Amendment No.: 225.

Facility Operating License No. NPF–6: Amendment authorized revision to the SAR and revised the TSs.

*Date of initial notice in **Federal Register:*** February 23, 2000 (65 FR 9006).

The June 29, 2000, supplement withdrew the proposed TS change to clarify the allowable containment leakage rate. The August 16, 2000, supplement withdrew the proposed TS change to increase the allowable containment spray pump degradation. The April 4, June 9, June 29, August 2, and August 16, 2000, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 29, 2000, as supplemented by letter dated October 4, 2000.

Brief description of amendment: The amendment revised the containment cooling system Technical Specifications to require that two independent containment cooling groups are operable with two operational cooling units in each group, in Modes 1, 2, 3, and 4.

Date of issuance: November 13, 2000.

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

Amendment No.: 226.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46008). The October 4, 2000, supplement provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 19, 2000, as supplemented July 19, 2000.

Brief description of amendments: The amendments will increase the setpoint tolerances for the pressurizer and main steam safety valves.

Date of Issuance: November 14, 2000.

Effective Date: November 14, 2000.

Amendment Nos.: 166 and 110.

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

*Date of initial notice in **Federal Register:*** April 5, 2000 (65 FR 17915). The July 19, 2000, submittal provided clarifying information that did not change the scope of the original **Federal Register** Notice or change the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 26, 2000.

Brief description of amendment: The amendment changes the Millstone Nuclear Power Station, Unit 3, Technical Specifications (TS) Section 1.13, Definitions, ''Engineered Safety Features Response Time''; TS Section 1.28, ''Reactor Trip System Response Time''; TS Section 3.3.1, ''Instrumentation-Reactor Trip System Instrumentation''; and TS Section 3.3.2, ''Instrumentation-Engineered Safety Features Actuation System Instrumentation'' to provide for verification of response time for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the Nuclear Regulatory Commission.

Date of issuance: November 3, 2000.

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

Amendment No.: 187.

Facility Operating License No. NPF–49: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48755).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 2000.

No significant hazards consideration comments received: No.

PECO Energy Company, PSEG Nuclear LLC, Delmarva Power and Light Company, and Atlantic City Electric Company

Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: November 17, 1999, as supplemented June 15, 2000.

Brief description of amendments: The amendments revise TS 5.5.7.c, ''Ventilation Filter Testing Program (VFTP)'' to include the requirement for laboratory testing of Engineered Safety Feature Ventilation System charcoal samples per American Society for Testing and Materials D3803–1989 and the application of a safety factor of 2.0 to the charcoal filter efficiency assumed in the plant design-basis dose analyses.

Date of issuance: November 3, 2000.

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

Amendments Nos.: 237 and 240.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4288). The June 15, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 3, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 12, 2000.

Brief description of amendment: The proposed amendment would revise Technical Specification 3/4.7.5, ''Ultimate Heat Sink,'' by increasing the minimum required service water pond level from 415 feet to 416.5 feet and decreasing the maximum allowed temperature at the discharge of the service water pumps from 95 degrees Fahrenheit to 90.5 degrees Fahrenheit.

Date of issuance: November 14, 2000.

Effective date: November 14, 2000.

Amendment No.: 149.

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

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46015).

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

No significant hazards consideration comments received: No.

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By December 29, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: November 2, 2000, as supplemented November 3, 2000.

Description of amendment request: The amendment revises the Technical Specifications to allow reactor coolant system (RCS) inservice leak and hydrostatic testing to be performed with the reactor in the cold shutdown mode while the RCS temperature is greater than 212 °F (which normally corresponds to the hot shutdown mode).

Date of issuance: November 3, 2000.

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

Amendment No.: 267.

Facility Operating License No. DPR–59: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated November 3, 2000.

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni.

Dated at Rockville, Maryland, this 21st day of November 2000.

For the Nuclear Regulatory Commission.

Suzanne C. Black,

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

[FR Doc. 00–30282 Filed 11–28–00; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 35–27280]

Filings Under the Public Utility Holding Company Act of 1935, as amended (''Act'')

November 21, 2000.

Notice is hereby given that the following filing(s) has/have been made with the Commission pursuant to provisions of the Act and rules promulgated under the Act. All interested persons are referred to the application(s) and/or declaration(s) for complete statements of the proposed transaction(s) summarized below. The application(s) and/or declaration(s) and any amendment(s) is/are available for public inspection through the Commission's Branch of Public Reference.

Interested persons wishing to comment or request a hearing on the application(s) and/or declaration(s) should submit their views in writing by December 18, 2000, to the Secretary, Securities and Exchange Commission, Washington, DC 20549–0609, and serve a copy on the relevant applicant(s) and/or declarant(s) at the address(es) specified below. Proof of service (by affidavit or, in the case of an attorney at law, by certificate) should be filed with the request. Any request for hearing should identify specifically the issues of facts or law that are disputed. A person who so requests will be notified of any hearing, if ordered, and will receive a