PM—Meeting Wrap-up/Future Business

Dated: September 30, 1999.

[FR Doc. 99–26007 Filed 10–5–99; 8:45 am]

BILLING CODE 7555–01–M

NATIONAL SCIENCE FOUNDATION

Special Emphasis Panel in Mathematical Sciences; Notice of Meeting

In accordance with the Federal Advisory Committee Act (Pub. L. 92–463, as amended), the National Science Foundation announces the following meeting:

Name: Special Emphasis Panel in mathematical Sciences (1204).

Date and Time: October 4 & 5, 1999.

Place: National Science Foundation.

Type of Meeting: Closed.

Contact Person: Hans Engler, Program Director, Applied Mathematics Program, or Joe Jenkins, Program Director, Analysis Program, Room 1025 National Science Foundation, 4201 Wilson Boulevard, Arlington, VA 22230, Telephone: (703) 306–1870.

Purpose of Meeting: To provide advice and recommendations concerning proposals submitted to NSF for financial support.

Agenda: To review and evaluate proposals concerning the Grants for Vertical Integration of Research and Education in the Mathematical Sciences (VIGRE) as part of the selection process for awards.

Reason for Closing: The proposals being reviewed include information of a proprietary of confidential nature, including technical information; financial data, such as salaries and personal information concerning individuals associated with the Proposals. These matters are exempt under 5 U.S.C. 552b(c)(4) and (6) of the Government in the Sunshine Act.

Dated: September 30, 1999.

Karen J. York,

Committee Management Officer.

[FR Doc. 99–25999 Filed 10–5–99; 8:45 am]

BILLING CODE 7555–01–M

NATIONAL SCIENCE FOUNDATION

Advisory Committee for Polar Programs: Notice of Meeting

In accordance with the Federal Advisory Committee (Pub. L. 92–463, as amended), the National Science Foundation announces the following meeting:

Name: Advisory Committee for Polar Programs (1130).

Date and Time: November 1, 1999—8:30 a.m. to 5 p.m, November 2, 1999—8:30 a.m. to 5 p.m.

Place: National Science Foundation, 4201 Wilson Blvd., Room 1235, Arlington, VA 22230.

Type of Meeting: Open.

Contact Person: Brenda Williams, Office of Polar Programs (OPP), National Science Foundation, 4201 Wilson Blvd., Suite 755, Arlington, VA 22230, (703) 306–1030.

Minutes: May be obtained from the contact person listed above.

Purpose of Meeting: To advise NSF on the impact of its policies, programs and activities on the polar research community; to provide advice to the Director of OPP on issues related to long range planning, and to form ad hoc sub-committees to carry out needed studies and tasks.

Agenda: Discussion of NSF-wide initiatives, long-range planning, and GPRA.

Dated: September 30, 1999.

Karen J. York,

Committee Management Officer.

[FR Doc. 99–26005 Filed 10–5–99; 8:45 am]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–445 and 50–446]

TXU Electric Co.; Notice of Issuance of Amendments to Facility Operating Licenses

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 72 to Facility Operating License No. NPF–87 and Amendment No. 72 to Facility Operating License No. NPF–89 issued to TXU Electric Company, which revised the Technical Specifications (TSs) for operation of the Comanche Peak Steam Electric Station (CPSES) located in Somervell County, Texas. The amendments are effective as of the date of issuance.

The amendments modified the rated thermal power (RTP), in paragraph 2.C.(1) of Facility Operating License No. NPF–89 (FOL NPF–89) for CPSES, Unit 2, from 3411 megawatts thermal (MWt) to 3445 MWt. The amendments also changed the TSs for CPSES, Units 1 and 2. The amendments changed TS 1.1 to increase the RTP to 3445 MWt for CPSES, Unit 2. In addition, the Allowable Values for the reactor trip setpoints for ''N–16 Overpower,'' and ''Power Range Neutron Flux—High'' in TS Table 3.3.1–1 are changed for CPSES, Unit 2 and TS 5.6.5b, ''Core Operating Limits Report (COLR),'' is changed to reflect appropriate, power-dependant, safety analysis assumptions and the updating of these assumptions in NRC staff-approved documents.

The application for the amendments 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 Amendments to Facility Operating Licenses and Opportunity for a Hearing in connection with this action was published in the **Federal Register** on May 10, 1999 (64 FR 25086). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the Environmental Assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (64 FR 43762).

For further details with respect to the action see (1) the application for amendments dated December 21, 1998, as supplemented by letters dated April 23, May 14, July 9, August 13 (two letters), August 25, and September 10, 1999, (2) Amendment No. 72 to License No. NPF–87, (3) Amendment No. 72 to License No. NPF–89, (4) the Commission's related Safety Evaluation, and (5) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the University of Texas at Arlington Library, Government Publications/Maps, 702 College, P. O. Box 19497, Arlington, Texas.

Dated at Rockville, Maryland, this 30th day of September, 1999.

For the Nuclear Regulatory Commission.

David H. Jaffe,

Senior Project Manager, Section 1, Project Directorate IV & Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–25975 Filed 10–5–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 11, 1999, through September 24, 1999. The last biweekly notice was published on September 22, 1999 (64 FR 51343).

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

By November 5, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: September 1, 1999.

Description of amendments request: The proposed amendment requests the following changes to the Technical Specifications:

1. Change the definition of Azimuthal Power Tilt in Technical Specification 1.1;

2. Correct the peak linear heat rate safety limit in Technical Specification 2.1.1.2;

3. Correct the DC voltage range listed in Surveillance Requirements 3.8.3.9 and 3.8.1.15;

4. Correct the loss of voltage and degraded voltage settings in Surveillance Requirement 3.3.6.2;

5. Correct the list of core operating limits in Technical Specification 5.6.5.a;

6. Correct a note on Technical Specification Figure 2.1.1–1;

7. Remove references to Unit 2, Cycle 12 in various Technical Specifications; and

8. Correct a typographical error in Technical Specification 5.6.

Specifically, the Proposed Technical Specifications are as follows:

1. Technical Specification 1.1 is proposed to be changed to replace the definition of Azimuthal Power Tilt with a new definition.

2. Technical Specification 2.1.1.2 is proposed to be changed by replacing the peak linear heat rate safety limit with less than or equal to 22kW/ft.

3. Technical Specification SR 3.3.6.2 is proposed to be changed by replacing the degraded voltage function with transient degraded voltage and steady-state degraded voltage functions.

4. Technical Specification SRs 3.8.1.9 and 3.8.1.15 are proposed to be changed by replacing the steady-state voltage range with the range of greater than or equal to 4060 volts and less than or equal to 4400 volts.

5. Technical Specification 5.6.5.a is proposed to be changed by adding Technical Specifications 3.1.4 and 3.3.1 to the list.

6. Technical Specification Figure 2.1.1–1 is proposed to be changed by removing the reference to Figure B2.1–1.

7. Various Technical Specifications and Figure 2.1.1–1a.

8. Technical specification 5.6.5.b, Item 41.ii is proposed to be changed by correcting CEN–199(B)–P to CEN–119(b)–P.

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

Change the Definition of Azimuthal Power Tilt

In their Infobulletin 97–07, Revision 1, Asea Brown Boveri, Inc.,—Combustion Engineering, Inc. (ABB–CE) stated that they had found a discrepancy in the Technical Specification definition of azimuthal power tilt. This discrepancy was found to exist in all CE Nuclear Steam supply System analog plants that use CECOR for monitoring and surveillance, and that use ABB–CE safety analysis methodology. Calvert Cliffs is one of those plants.

The value of Tq (Azimuthal tilt magnitude) as used in the azimuthal power tilt formula now in Technical Specification 1.1 is not conservative in all cases. With the proposed definition, Tq is the maximum fractional increase in power that can occur anywhere in the core because of tilt. Since Tq is the maximum value, it is consistently conservative. This is the appropriate measured value of tilt to be used in verifying that the tilt assumed in establishing safety limits has not been exceeded.

Therefore, changing the definition of azimuthal power tilt as proposed will not involve a significant increase in the probability of consequences of an accident previously evaluated.

Correct the Peak Linear Heat Rate Safety Limit

When Improved Standard Technical Specifications (ITS) were written, the peak linear heat rate safety limit of [less than or equal to] 21 kW/ft was inadvertently written in Technical specification 2.1.1.2. the correct number is [less than or equal to] 22kW/ft. the peak linear heat rate safety limit was established at [less than or equal to] 22 kW/ft in License Amendment Nos. 88 (Unit 1) and 61 (Unit 2). This number was valid for both units at the time of implementation of ITS.

Therefore, changing the peak linear heat rate safety limit to a number previously approved by the Nuclear Regulatory Commission (NRC) will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Correct the Diesel Generator Loss of Voltage and Degraded Voltage Settings

When the ITS were written, a single set of numbers for the degraded voltage function was provided in Technical Specification Surveillance Requirement (SR) 3.3.6.2. The degraded voltage function should have been expressed as transient degraded voltage and steady-state degraded voltage. This separation of two types of degraded voltage functions was approved in License Amendment Nos. 226 (Unit 1) and 200 (Unit 2), which were issued before the ITS were approved.

Therefore, changing the degraded voltage function to the transient degraded voltage and steady-state degraded voltage functions previously approved by the NRC will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Correct the Diesel Generator Voltage Range

Technical Specification SRs 3.8.1.9 and 3.8.1.15 require each diesel to be started from a stand-by condition. Surveillance requirement 3.8.1.9 requires that the generator reach [greater than or equal to] 3740 volts within 10 seconds. After steady-state conditions are reached, both SRs require the generator to maintain a voltage range of greater than 3740 volts and [less than or equal to] 4580 volts.

The Baltimore Gas and Electric Company ITS conversion added voltage requirements to SRs 3.8.1.9 and 3.8.1.15 consistent with SR 3.8.1.3. License Amendment Nos. 226 and 200 changed the voltage requirement for SR 3.8.1.3 to [greater than or equal to] 4060 volts and [less than or equal to] 4400 volts. The voltage was not corrected in SRs 3.8.1.9 and 3.8.1.15 when the Technical Specifications were changed to ITS.

Therefore, changing the voltage in SRs 3.8.1.9 and 3.8.1.15 to voltage previously approved by the NRC will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Correct the List of Core Operating Limits

Technical Specification 5.6.5.a lists Technical Specifications that are to be included in the core operating limits and documented in the Core Operating Limits Report (COLR). In the transition to ITS, Technical Specifications 3.1.4 (Control Element Assembly Alignment) and 3.3.1 (Reactor Protective System—Operating) were inadvertently omitted from the list. The complete list is currently in the COLR.

Therefore, restoring Technical Specification 5.6.5.a to a list previously approved by the NRC will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Correct Figure 2.1.1–1

A note of Technical Specification Figure 2.1.1–1 was changed in License Amendment Nos. 227 (Unit 1) and 201 (Unit 2) (ITS) to delete reference to Figure B2.1–1. Figure B2.1–1 was deleted from the Technical Specification Bases in the transition to ITS. In License Amendment Nos. 228 (Unit 1) and 202 (Unit 2), an old version of Figure 2.1.1–1 was used, and the reference to Figure B2.1–1 was thus inadvertently put back in the note. The proposed correction will replace the reference to Figure B2.1–1 with the wording approved in License Amendment Nos. 227 and 201.

Therefore, returning the note in Figure 2.1.1–1 to the wording previously approved by the NRC will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Remove References to Unit 2, Cycle 12

License Amendment Nos. 228 and 202 added notes to indicate areas in the Technical Specifications that had special application to Cycle 12 of Unit 2 only. Cycle 12 of Unit 2 ended in May 1999. Since these notes no longer have application, they are proposed to be removed. Additionally, Figure 2.1.1-la applies only to Unit 2, Cycle 12, and it is proposed to be removed.

Therefore, removal of information no longer applicable to either unit is an administrative change and will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Correct a Typographical Error

Technical Specification 5.6.5.b, Item 41.ii is being corrected to change the number of the publication ''BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2'' from CEN–199(B) to CEN–119(B)–P. Correction of a typographical error does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from an accident previously evaluated.

Change the Definition of Azimuthal Power Tilt

In their Infobulletin 97–07, Revision 1, ABB–CE stated that they had found a discrepancy in the Technical specification definition of azimuthal power tilt. This discrepancy was found to exist in all CE Nuclear Steam Supply System analog plants that use CECOR for monitoring and surveillance and that use ABB–CE safety analysis methodology. Calvert Cliffs is one of those plants.

The value of Tq (azimuthal tilt magnitude) as used in the azimuthal power tilt formula now in Technical specification 1.1 is not always the most conservative in all cases. With the proposed definition, Tq is the maximum fractional increase in power that can occur anywhere in the core because of tilt. Since Tq is the maximum value, it is conservative. This is the appropriate measured value of tilt to be used in verifying that the tilt assumed by ABB–CE in establishing safety limits has not been exceeded.

Therefore, changing the definition of azimuthal power tilt as proposed will not create the possibility of a new or different type of accident from any accident previously evaluated.

Correct the Peak Linear Heat Rate

When the ITS were written, a value of peak linear heat rate [less than or equal to] 21 kW/ft was inadvertently written in Technical Specification 2.1.1.2. The correct number is [less than or equal to] 22 kW/ft. The required peak linear heat rate was established at [less than or equal to] 22 kW/ft in License Amendment Nos. 88 and 61. This number was valid for both units at the time of implementation of ITS.

Therefore, changing the value of peak linear heat rate to a value previously approved by the NRC will not create the possibility of a new or different type of accident from any accident previously evaluated.

Correct the Diesel Generator Loss of Voltage and Degraded Voltage Settings

When the ITS were written, a single set numbers for the degraded voltage function was provided in Technical specification SR 3.3.6.2. The degraded voltage function should have been expressed as transient degraded voltage and steady-state degraded voltage. This separation of two types of degraded voltage functions was approved in License Amendment Nos. 226 and 200, which were issued before the ITS were approved.

Therefore, changing the degraded voltage function to the transient degraded voltage and steady-state degraded voltage functions previously approved by the NRC will not create the possibility of a new or different type of accident from any accident previously evaluated.

Correct the Diesel Generator Voltage Range

Technical Specification SRs 3.8.1.9 and 3.8.1.15 require that each diesel be started from a stand-by condition. Surveillance Requirement 3.8.1.9 requires that the generator reach [greater than or equal to] 3740 volts within 10 seconds. After steady-state conditions are reached, both SRs require the generator to maintain a voltage range of greater than 3740 volts and [less than or equal to] 4580 volts.

The Baltimore Gas and Electric Company ITS conversion added voltage requirements to SRs 3.8.1.9 and 3.8.1.15 consistent with SR 3.8.1.3. License Amendment Nos. 226 and 200 changed the voltage requirement for SR 3.8.1.3 to [greater than or equal to] 4060 volts and [less than or equal to] 4400 volts. The voltage was not corrected in SRs 3.8.1.9 and 3.8.1.15 when the Technical Specifications were changed to ITS.

Therefore, changing the voltage in SRs 3.8.1.9 and 3.8.1.15 to a voltage previously approved by the NRC will not create the possibility of a new or different type of accident from any accident previously evaluated.

Correct the List of Core Operating Limits

Technical Specification 5.6.5.a lists Technical specifications that are to be included in the core operating limits and documented in the COLR. In the transition to ITS, Technical Specifications 3.1.4 (Control Element Assembly Alignment) and 3.3.1 (Reaction Protective System—Operating) were inadvertently omitted from the list. The complete list is currently in the COLR.

Therefore, restoring Technical Specification 5.6.5.a to a list previously approved by the NRC will not create the possibility of a new or different type of accident from any accident previously evaluated.

Correct Figure 2.1.1–1

A note on Technical Specification Figure 2.1.1–1 was changed in License Amendment Nos. 227 and 201 (ITS) to delete reference to Figure B2.1–1. Figure B2.1–1 was deleted from the Technical Specification Bases in the transition of ITS. In License Amendment Nos. 228 and 202, an old version of Figure 2.1.1–1 was used, and the reference to Figure B2.1–1 was thus inadvertently put back in the note. The proposed correction will replace the reference to Figure B2.1–1 with the wording approved in License Amendment Nos. 227 and 201.

Therefore, removal of information no longer applicable to either unit is an administrative change and will not create the possibility of a new or different type of accident from any accident previously evaluated.

Remove References to Unit 2, Cycle 12

License Amendment Nos. 228 and 202 added notes to indicate areas in the Technical Specifications that had special application to Cycle 12 of Unit 2 only. Cycle 12 of Unit 2 ended in May 1999. Since these notes no longer have application, they are proposed to be removed. Additionally, Figure 2.1.1–1a applies only to Unit 2, Cycle 12, and is proposed to be removed.

Therefore, removal of information no longer applicable to either unit is an administrative change and will not create the possibility of a new or different type of accident from any accident previously evaluated.

Correct a Typographical Error

Technical Specification 5.6.5.b, Item 41.ii is being corrected to change the number of the publication ''BASSS, Use of the Incore Detector System to Monitor the DNB–LCO on Calvert Cliffs Unit 1 and Unit 2'' from CEN–199(B)–P to CEN–119(B)–P. Correction of a typographical error will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

Change the Definition of Azimuthal Power Tilt

The margin of safety in this case is whether the azimuthal power tilt calculation shows the highest (most conservative) value for Tq (azimuthal tilt magnitude).

The value of Tq as used in the azimuthal power tilt formula now in Technical Specification 1.1 is not always the most conservative in all cases. With the proposed definition, Tq is the maximum fractional increase in power that can occur anywhere in the core because of tilt. Since Tq is the maximum value, it is conservative. This is the appropriate measured value of tilt to be used in verifying that the tilt assumed in establishing safety limits has not been exceeded.

Therefore, changing the definition of azimuthal power tilt as proposed will not involve a significant reduction in the margin of safety.

Correct the Peak Linear Heat Rate Safety Limit

The margin of safety in this case was previously approved by the NRC in License Amendment Nos. 88 and 61.

Correct the Diesel Generator Loss of Voltage and Degraded Voltage Settings

The margin of safety in this case was previously approved by the NRC in License Amendment Nos. 226 and 200.

Correct the Diesel Generator Voltage Range

The margin of safety in this case was previously approved by the NRC in License Amendment Nos. 226 and 200.

Correct the List of Core Operating Limits

Technical Specification 5.6.5.a lists Technical specifications that are to be included in the core operating limits and documented in the COLR. In the transition to ITS, Technical Specifications 3.1.4 (Control Element Assembly Alignment) and 3.3.1 (Reactor Protective System—Operating) were inadvertently omitted from the list. The complete list is currently in the COLR.

Therefore, restoring Technical Specification 5.6.5.a to a list previously approved by the NRC will not involve a significant reduction in the margin of safety.

Correct Figure 2.1.1–1

A note on Technical Specification Figure 2.1.1–1 was changed in License Amendment Nos. 227 and 201 (ITS) to delete reference to Figure B2.1–1. Figure B2.1–1 was deleted from the Technical Specification Bases in the transition to ITS. In License Amendment Nos. 228 and 202, an old version of figure 2.1.1–1 was used, and the reference to Figure B2.1–1 was thus inadvertently put back in the note. The proposed correction will replace the reference to Figure B2.1–1 with the wording approved in License Amendment Nos. 227 and 201.

Therefore, returning the note in Figure 2.1.1–1 to the wording previously approved by the NRC will not involve a significant reduction in the margin of safety.

Remove References to Unit 2, Cycle 12

License Amendment Nos. 228 and 202 added notes to indicate areas in the Technical Specifications that had special application to Cycle 12 of Unit 2 only. Cycle 12 of Unit 2 ended in May 1999. Since these notes no longer have application, they are proposed to be removed. Additionally, Figure 2.1.1–1a applies only to Unit 2, Cycle 12, and it is proposed to be removed.

Therefore, removal of information no longer applicable to either unit is an administrative change and will not involve a significant reduction in the margin of safety.

Correct a Typographical Error

Technical specification 5.6.5.b, Item 41.ii is being corrected to change the number of the publication ''BASSS, Use of the Incore Detector system to Monitor the DNB–LCO on Calvert cliffs Unit 1 and Unit 2'' from CEN–199(B)–P to CEN–119(B)–P. Correction of a typographical error will not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

NRC Section Chief: S. Singh Bajwa.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: August 26, 1999.

Description of amendment request: The proposed amendment would revise TS 3/4.9.4, ''Containment Building Penetrations,'' and its associated Bases to allow penetrations which provide direct access from the containment atmosphere to the outside atmosphere to remain open during refueling operations provided certain administrative controls are met.

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

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

Containment is not an accident initiating system as described in the Final Safety Analysis Report. This change is applicable only in Mode 6 during Core Alterations or movement of irradiated fuel (which occurs when the unit is shutdown). The proposed change will not modify equipment used for fuel movement or core alterations within the HNP [Harris Nuclear Plant] Containment Building. Administrative controls will be used to isolate containment in the event of a fuel handling accident. The consequences of a Fuel Handling Accident inside containment will increase as a result of this change. However, the proposed administrative controls will require closure of containment prior to exceeding standard review plan dose limits due to a radiological release from a design basis fuel handling accident.

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

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

The proposed change provides for administrative controls and operating restrictions for air lock doors consistent with previous guidance authorized by the Commission for similar nuclear power plants. Containment is not an accident initiating system as described in the Final Safety Analysis Report. Fuel Handling Accidents have been previously analyzed for the Harris Nuclear Plant.

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

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

Administrative controls will be used to isolate containment in the event of a fuel handling accident. The proposed administrative controls will require closure of containment prior to exceeding standard review plan dose limits due to a radiological release from a design basis fuel handling accident.

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

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

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 29, 1999.

Description of amendment request: The proposed change to the Arkansas Nuclear One, Unit 2, Technical Specifications would allow the performance of a special inspection of the steam generator tubes during an upcoming mid-cycle outage. This mid-cycle outage is planned for the purpose of performing inspections in selected areas of the steam generator tube bundle where previous inspections have revealed tube degradation. The proposed change would limit the initial inspection scope to these identified areas and includes a scope expansion criteria to address unexpected conditions.

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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

This change has no actual impact on any previously analyzed accident in the final safety analysis report (FSAR). A double-ended break of one steam generator tube is postulated as part of the ANO–2 design basis accident evaluation. The change permits Entergy Operations to determine the appropriate scope and expansion criteria for a special steam generator tube inspection that is being performed at a frequency more conservative than that of the augmented inservice inspection program included in the TSs [Technical Specifications]. The special inspection will find and repair certain steam generator tubing flaws that would otherwise remain in service until the next scheduled refueling outage. The increased inspection frequency reduces the probability that a flaw in a steam generator tube could grow to a size that would affect the leakage or structural integrity of the tube. The augmented inservice inspection program contained in the TSs is not being modified.

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

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

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

The scope of this change does not establish a potential new accident precursor. The design basis accident analyses for ANO–2 include the consequences of a double-ended break of one steam generator tube which bounds other postulated failure mechanisms. The proposed change would permit determination of alternate inspection criteria for a special inspection which is in addition to the periodic inservice inspections required by the TSs. The equipment used in the special inspection would not affect any plant components differently than those used for current TS required inspections.

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

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

As previously stated, a double-ended rupture of one steam generator tube is accounted for in the ANO–2 design basis accident analysis. Considering that the 2P99 special inspection is in addition to the inservice inspection program defined in the ANO–2 TSs and that leakage detection capability is not being modified, performance of a special inspection of any scope will increase the margin of safety over the current TS requirements.

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

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 17, 1999.

Description of amendment request: The proposed amendment would remove the voltage-based repair criteria, F* repair criteria, and sleeving methodologies from the Unit 1 Technical Specifications (T/S) and clarify the Bases sections accordingly.

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

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

This change removes the interim steam generator tube plugging criteria from the T/S and reinstates the original T/S criteria consistent with Unit 2 (which does not have significantly degraded steam generators). The current T/S allow for continued operation with tubes that demonstrate indications per F* and voltage-based criteria. The basis used to justify the interim criteria is specific to the Unit 1 original steam generators (OSGs) and does not apply to the replacement steam generators (RSGs).

The proposed change returns the plugging criteria for the steam generator tubes to the original licensing basis. The criteria are in accordance with NUREG–0452, (old) ''Standard Technical Specifications.'' The plugging criteria are based on a minimum wall thickness due to wastage as determined by ASME [American Society of Mechanical Engineers] Section XI. The proposed change is conservative in nature because it does not allow for continued operation with F* and voltage-based degraded tubes. Because of this, the probability of a steam generator tube rupture (SGTR) is not increased.

The potential for a SGTR is also not increased as demonstrated in the qualification analysis and testing for the RSGs. The program for periodic in-service inspection monitors the integrity of the SG tubing to provide reasonable assurance that there is sufficient time to take proper and timely corrective action if any tube degradation is detected. The tube inspections themselves are not initiators of a SGTR. Therefore, this change is not expected to increase the probability of a SGTR during normal or accident conditions.

Unit 1 will continue to apply the T/S maximum primary-to-secondary leakage limit of 150 gallons per day (gpd) through any one SG to minimize the potential for excessive leakage. The EPRI [Electric Power Research Institute]-recommended 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected tube leak and minimizes the potential for excessive leakage or tube burst in the event of main steamline break (MSLB) or loss-of-coolant accident (LOCA) conditions. This lower limit is more restrictive than the limit (500 gpd per SG and total leakage of 1440 gpd) utilized for determination of offsite dose and also provides further assurance that the probability of a SGTR is not increased.

The design basis doses calculated for postulated accidents involving degradation of SG tubes, such as SGTR and MSLB accidents, as presented in UFSAR chapter 14 accident analysis, have been evaluated. The SGTR consequences continue to be bounded by the design basis analyses due to the allowable leakage rate specified by this change. The proposed T/S leakage rate is maintained at 150 gpd per SG. However, the maximum leakage of 500 gpd per SG and total leakage of 1440 gpd for all four generators was used to determine offsite dose in UFSAR chapter 14. The MSLB consequences are decreased by installation of the RSGs due to the reduction in primary-to-secondary leakage during the MSLB. Under the approved interim plugging criteria, a leak rate of 8.4 gpm was determined to be the upper limit for allowable primary-to-secondary leakage in the faulted steam generator. This leakage, combined with the 150 gpd leakage from the non-faulted SGs, was determined to limit the offsite dose to 10% of the 10 CFR 100 limits. Following replacement of the SGs, the leakage is limited during the MSLB to 150 gpd for both the faulted and unfaulted SGs. Therefore, the Unit 1 MSLB dose will be bounded by the current Unit 2 dose analysis, which is less than 10% of 10 CFR 100 limits.

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

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

Removing application of voltage-based repair criteria, F* repair criteria, and sleeving methodologies upon installation of the RSGs will not introduce significant or adverse changes to the plant design basis that could lead to a new or different kind of accident being created. This change does not change the overall objective of surveillance activities—maintaining the structural integrity of this portion of the reactor coolant system. The surveillance activities are performed during outages. The proposed change in the surveillance program returns the program to the initial licensing basis. No new failures are created.

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

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

Removing the application of voltage-based and F* repair criteria and sleeving methodologies does not involve a reduction in the margin of safety. The RSG tubing has been shown to retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions consistent with GDC 14, 15, 30, 31, and 32 of 10 CFR [Part] 50 [A]ppendix A. The RSG tubing has been designed and evaluated consistent with the ASME Section III, 1989 edition. The proposed plugging criteria are based on ASME Section XI and do not allow for operation with indications identified by F* and voltage-based criteria. The proposed program for periodic in-service inspection of the RSGs monitors the integrity of the SG tubing to provide reasonable assurance that there is sufficient time to take proper and timely corrective action if any tube degradation is present. The proposed program is consistent with NUREG–0452 and was the basis for the original Unit 1 T/S surveillance program.

The proposed change maintains the T/S maximum primary-to-secondary leakage at 150 gpd per generator to minimize the potential for excessive leakage. This limit provides for leakage detection and shutdown in the event of an unexpected tube leak and minimizes the potential for excessive leakage or tube burst in the event of a MSLB or LOCA. Because this limit is maintained, the margin of safety is maintained.

Therefore, it is concluded that 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.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

Attorney for licensee: Jeremy J. Euto, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: September 10, 1999.

Description of amendment requests: The proposed amendments would revise Technical Specification (T/S) 3/4.4.7 so that the surveillance requirement does not need to be performed when the reactor is defueled with no forced circulation. The proposed revision to T/S 3/4.4.7 also includes changes to Tables 3.4–1 and 4.4–3. A change is proposed to Unit 1 T/S Table 4.4–3 to revise the reactor coolant system (RCS) chemistry sampling frequency from three times per 7 days with a maximum interval of 72 hours to a frequency of at least once per 72 hours. An editorial change to Unit 1 Tables 3.4–1 and 4.4–3 would relocate the asterisk for the footnote to a position adjacent to the parameter ''dissolved oxygen,'' from its current position next to the allowable chemistry limit in Table 3.4–1 and the analysis frequency in Table 4.4–3. An editorial change would also correct the footnote for Table 3.4–1 for Unit 1 and Unit 2 by making the word ''limit'' plural, as it applies to both the steady-state and transient limits.

Changes are also proposed to revise Surveillance Requirement 4.11.2.2 by deleting the phrase ''by analysis of the Reactor Coolant System noble gases.''

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

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

The proposed changes to the RCS chemistry sampling requirements do not affect the probability of a loss-of-coolant accident or steam generator tube rupture, which are evaluated in Sections 14.3 and 14.2.4, respectively, of the Updated Final Safety Analysis Report (UFSAR). RCS contaminant limits are maintained to reduce the potential for RCS leakage or failure due to corrosion. Sampling the RCS for contaminants does not initiate an accident. Deleting the requirement to obtain samples when the reactor is defueled does not modify any plant equipment or affect plant operation and therefore does not introduce any new accident initiators or precursors. Suspension of RCS chemistry sampling when the reactor is defueled does not increase the potential for RCS leakage or failure because the corrosive effects of the contaminants is minimal during this low-temperature, low-pressure condition. To ensure elevated contaminant levels would be detected and corrected prior to subjecting the system to a high-temperature condition, chemistry sampling will be reinstated within 72 hours of re-establishing forced circulation and prior to entering Mode 6. Removing the restriction for analyzing primary coolant chemical contaminants at least three times every seven days does not change the maximum surveillance interval. This change allows the sample to be collected two or three times per week, consistent with the maximum 72-hour interval. The 72-hour sampling and analysis interval is consistent with the current requirement in the Unit 2 T/S, and industry guidance in NUREG–0452, ''Standard Technical Specifications.'' The 72-hour interval continues to provide adequate assurance that concentrations in excess of the limits are detected in sufficient time to take corrective actions. Therefore, the probability of occurrence of a previously evaluated accident is not increased.

This change does not alter the quantity of radioactive material in any system during normal plant operation, the amount of shielding provided by plant systems, or the mitigative capabilities of any system following an event. Therefore, the consequences of a previously evaluated accident are not increased.

The editorial changes to the RCS chemistry T/S provide consistency between the Unit 1 and Unit 2 T/S and the Standard Technical Specifications. These changes do not affect the design or operation of any system, structure, or component in the plant. The accident analysis assumptions and results are unchanged. No new failures or interactions are created.

The amount of radioactive material in the gas storage tanks is controlled to ensure that, in the event of a rupture of one of these tanks, the resulting total body exposure to an individual at the nearest site boundary would not exceed 0.5 rem. The accidental waste gas release event is summarized in Section 14.2.3 of the UFSAR. Sampling to determine the radioactivity levels in the tanks does not initiate an accident or identify any accident precursors. The increased sampling flexibility does not change the method of operating the waste gas system, nor does it modify any interfaces with other plant systems. Therefore, this change does not increase the probability of occurrence of an accidental waste gas release event.

Implementation of a different sampling method does not change the maximum quantity of radioactive material specified in the T/S Limiting Condition for Operation (LCO). The sampling method has no effect on normal plant gaseous radwaste activities, so the composition of the radioactive gaseous nuclides present in the tank at the time of the event is not affected. As the proposed revision allows a change to the method of sampling but does not affect the radioactivity limit for the gas storage tanks, the proposed change does not increase the consequences of an accidental waste gas release event.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not increased.

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

The proposed changes to revise the RCS chemistry sampling frequency and to suspend RCS chemistry sampling when the reactor is defueled with no forced circulation does not change the method of operating any equipment or the operational limits of any equipment. The proposed changes do not introduce any new failure mechanisms to the RCS or any other plant systems. The proposed change does not involve any physical alterations to any plant equipment, and causes no change in the method by which any plant system performs its function. Editorial changes to footnotes for Tables 3.4–1 and 4.4–3 provide consistency between the T/S for Unit 1 and Unit 2, but do not change the methods of operating any equipment or introduce any new failure mechanisms.

The proposed change to eliminate the prescriptive waste gas tank sampling method does not introduce any new failure mechanisms to the waste disposal system, involve any physical changes to the waste disposal system or any other plant systems, or change the way any plant systems are operated. This change does not change any interfaces between the waste disposal system and any other plant systems. The proposed changes continue to ensure the system is operated within the existing limit established by the T/S LCO. Thus, no adverse safety considerations are introduced by this proposed change to the T/S.

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

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

The margin of safety pertinent to the RCS chemistry surveillance is related to the concentration of chemical contaminants that would expedite corrosion of the RCS piping and components and the period of time during which the system is allowed to operate outside the T/S limits. The proposed changes to the RCS chemistry surveillance do not alter either of these criteria. These proposed changes do not affect any safety limits or T/S parameter limits. The proposed changes do not introduce new equipment, equipment modifications, or new or different modes of plant operation. These changes do not affect the operational characteristics of any equipment or systems. The editorial changes to footnotes for Tables 3.4–1 and 4.4–3 provide consistency between the T/S for Unit 1 and 2, but do not affect the acceptance criteria or surveillance frequencies for this T/S.

The margin of safety pertinent to the waste gas storage tanks is related to the quantity of radioactivity that would be released in the unlikely event of a tank rupture. The proposed change to the gas storage tank T/S eliminates the prescriptive sampling methodology, but does not affect the requirement to periodically quantify the radioactive gaseous material in the gas storage tanks. The proposed change does not affect the quantity of radioactivity allowed in the gas storage tanks, nor does it alter the methodology, assumptions, or results of any safety analyses. The proposed change to delete the prescriptive sampling method does not affect any safety limits or T/S parameter limits.

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 requests involve no significant hazards consideration.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

Attorney for licensee: Jeremy J. Euto, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

National Aeronautics Space Administration (NASA), Docket No. 50–30, NASA Test Reactor, Erie County, Ohio

Date of amendment request: March 25, 1999, as supplemented by letter dated August 10, 1999.

Description of amendment request: The proposed amendment would change Lewis Research Center (LeRC) to Glenn Research Center (GRC).

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

The proposed amendment will change the name of the Licensee for the Plum Brook Reactor Facility (PBRF) TR–3 license, a possession only license, from Lewis Research Center (LeRC) to the Glenn Research Center (GRC). The amendment request is necessary because NASA has changed the name of the Lewis Research Center to the Glenn Research Center at Lewis Field under legislative action and signed into law (sec. 434, P.L. 105–276, 112 Stat. 2461) on October 21, 1998. The effective date of this name change was March 1, 1999. NASA, GRC will retain the PBRF license and the responsibility to continue maintaining the PBRF Reactor Facility in a safe protected storage mode under the current TR–3 possess-but-not-operate license. In addition, the current plans to provide a PBRF decommissioning plan to the NRC by the end of CY 1999 and the eventual decommissioning by the end of CY 2007 have not changed.

There will be no change in the funding status of the GRC in either maintaining the PBRF facility in the safe protected storage mode or the eventual decommissioning. NASA, as a government agency, remains responsible for the continuing funding of both activities.

In addition, there will be no change in the personnel who are responsible for maintaining the present TR–3 license or in developing the PBRF Decommissioning Plan.

The proposed amendment does not require any physical change to the PBRF Facility, changes to the Technical Specifications or procedures under the PBRF TR–3 License other than the name change from LeRC to GRC. The proposed change does not increase the probability of any accident or increased risk to the public safety.

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

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

The proposed amendment does not modify the PBRF facility configuration or licensed activities. Therefore, no additional accident conditions are introduced.

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

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

This amendment is required because of the name change from LeRC to GRC. NASA will continue to be financially responsible to maintain the PBRF Facility under the existing TR–3 License.

Furthermore, the GRC personnel for the eventual PBRF decommissioning and contract support personnel reporting to GRC will continue to be technically qualified to maintain the PBRF under the safe protected storage mode. There has been no effective change in the personnel who will be responsible to implement the eventual decommissioning effort that will be required under the future PBRF Decommissioning Plan.

Plum Brook's existing qualified contractors remained in place following the name change. The requested amendment does not involve any changes in the performance of current licensed activities and these activities will continue in their current form without changes or interruptions of any kind.

The proposed amendment does not alter any margin of safety because it does not involve any changes in the PBRF Facility or licensed activities under the TR–3 License. All activities will continue in the current form without changes or interruptions of any kind as a result of the name.

Therefore, the proposed amendment 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Local Public Document Room location: N/A.

Attorney for licensee: Elias T. Naffah, MS 500–118, NASA, Glenn Research Center, 21000 Brookpark Road, Cleveland Ohio 44135.

NRC Branch Chief: Ledyard B. Marsh.

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

Date of amendment request: July 16, 1999.

Description of amendment request: Proposed relocation of Technical Specifications 3/4.9.3.2, ''Refueling Operations, Spent Fuel Temperature,'' 3/4.9.3.3, ''Refueling Operations, Decay Time,'' 3/4.9.5, ''Refueling Operation, Communications,'' 3/4.9.6, ''Refueling Operation, Crane Operability—Containment Building,'' and 3/4.9.7, ''Refueling Operations, Crane Travel—Spent Fuel Storage Building,'' to the Millstone, Unit No. 2 Technical Requirements Manual. The associated Bases pages and index pages will be modified to address the proposed change.

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 3/4.9.3.2, ''Refueling Operations, Spent Fuel Pool Temperature,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. This specification limits spent fuel pool temperature to be less than or equal 140 °F to ensure the resin in the spent fuel cooling demineralizers will not degrade and the temperature and humidity are compatible with personnel comfort and safety requirements. Additionally, the requirement ensures that the design temperature of the fuel pool cooling system, liner/building structures, and racks is not exceeded. Relocation of this Technical Specification to the TRM does not imply any reduction in its importance in limiting the spent fuel pool bulk temperature to be less than or equal to 140 °F. Spent fuel pool bulk temperature is a design bases process variable which is used to establish the required heat removal capabilities of the spent fuel heat removal system. In the unlikely event of total loss of cooling water flow to the spent fuel pool, the pool water temperature may reach 212 °F within approximately 9 hours and will result in a boiling condition. This event does not represent a challenge to the fuel cladding, as a fission product barrier, unless the fuel becomes uncovered. The requirement on storage pool water level is covered by Technical Specification 3/4.9.12, ''Storage Pool Water Level,'' which requires a minimum of 23 feet of water over the top of irradiated fuel assemblies. Therefore, spent fuel pool bulk temperature is not by itself a process variable that is an initial condition of a design basis accident. This Technical Specification does not cover a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. It does not cover a structure, system, or component that is part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The proposed change will not alter the way pool temperature is measured, nor will it alter any of the assumptions used in the spent fuel pool fuel handling accident analysis. Relocation of this Technical Specification to the TRM does not degrade the performance of any safety systems or prevent actions assumed in the accident analysis, nor does it alter any of the assumptions made in the analysis that could increase the consequences of accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3/4.9.3.3, ''Refueling Operations, Decay Time,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. This specification requires the reactor to remain in Mode 5 or 6 until the most recent core offload has decayed a sufficient time to ensure alternate cooling is available during this time to cool the spent fuel pool should a failure occur in the Spent Fuel Pool Cooling System. Alternate cooling would be provided by the Shutdown Cooling System. Relocation of this Technical Specification to the TRM does not imply any reduction in its importance in insuring that the most recent core offload has decayed a sufficient time. If the requirement to remain in Mode 5 or 6 until the most recent core offload has decayed for 504 hours is not satisfied, the spent fuel pool cooling system may not have the capability to remove decay heat and stay below the Technical Specification limit of 140 °F. In the unlikely event of total loss of cooling water flow to the spent fuel pool, the pool water temperature may reach 212 °F in less than 9 hours and will result in a boiling condition. This event does not represent a challenge to the fuel cladding, as a fission product barrier, unless the fuel becomes uncovered. The requirements on storage pool water level is covered by Technical Specification 3/4.9.12, ''Storage Pool Water Level,'' which requires a minimum of 23 feet of water over the top of irradiated fuel assemblies. Therefore, this requirement to remain in Mode 5 or 6 until the most recent core offload has decayed for 504 hours is not by itself a process variable that is an initial condition of a design basis accident. This Technical Specification does not cover a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. It does not cover a structure, system, or component that is part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The proposed change will not alter the requirement that the most recent core offload has decayed a sufficient time, nor will it alter any of the assumptions used in the spent fuel pool fuel handling accident analysis. Relocation of this Technical Specification to the TRM does not degrade the performance of any safety systems or prevent actions assumed in the accident analysis, nor does it alter any of the assumptions made in the analysis that could increase the consequences of accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3/4.9.5, ''Refueling Operations, Communications,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. This specification requires communication between the control room and the refueling station, to ensure any abnormal change in the facility status, as indicated on the control room instrumentation, can be communicated to the refueling station personnel. Relocation of this Technical Specification to the TRM does not imply any reduction in its importance in insuring communication between the control room and the refueling station. This Technical Specification does not cover a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. It does not cover a structure, system, or component that is part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The proposed change will not alter the requirement on communication between the control room and the refueling station, nor will it alter any of the assumptions used in the spent fuel pool fuel handling accident analysis. Relocation of this Technical Specification to the TRM does not degrade the performance of any safety systems or prevent actions assumed in the accident analysis, nor does it alter any of the assumptions made in the analysis that could increase the consequences of accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3/4.9.6, ''Refueling Operations, Crane Operability—Containment Building,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. This specification ensures the lifting device on the refueling machine has adequate capacity to lift the weight of a fuel assembly and a control element assembly, and that an automatic load limiting device is available to prevent damage to the fuel assembly during fuel movement. Relocation of this Technical Specification to the TRM does not imply any reduction in its importance in insuring that the lifting device on the refueling machine has adequate capacity. The automatic load limiting device and/or physical stops are not monitored and controlled during operation, nor are they assumed to function to mitigate the consequences of a design basis accident. The automatic load limiting device is checked on a periodic basis to ensure operability. This Technical Specification, which ensures the lifting device on the refueling machine has adequate capacity, does not cover a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The proposed change will not alter the requirement that the lifting device on the refueling machine has adequate capacity, nor will it alter any of the assumptions used in the accident analysis. Relocation of this Technical Specification to the TRM does not degrade the performance of any safety systems or prevent actions assumed in the accident analysis, nor does it alter any of the assumptions made in the analysis that could increase the consequences of accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3/4.9.7, ''Refueling Operations, Crane Travel—Spent Fuel Storage Pool Building,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. This specification ensures loads in excess of one fuel assembly containing a control element assembly, plus the weight of the fuel handling tool, will not be moved over other fuel assemblies in the spent fuel storage racks. Therefore, in the event of a drop of this load, the activity released is limited to that contained in one fuel assembly. Relocation of this Technical Specification to the TRM does not imply any reduction in its importance in insuring that loads in excess of 1800 pounds (except of a consolidated fuel storage box) are prohibited from travel over irradiated fuel. While this Technical Specification does address an operating restriction assumed in the accident analysis, there is no process variable that can be monitored during power operation of the plant. Crane interlocks and/or physical stops are used to assure that this requirement is met, but indication of the operation of the interlocks and/or physical stops is not available in the control room. These features inhibit movement of the crane so that monitoring is not necessary. This Technical Specification does not cover a structure, system, or component that is part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The proposed change will not alter the requirement that the crane interlocks and/or physical stops are OPERABLE, nor will it alter any of the assumptions used in the spent fuel pool fuel handling accident analysis. Relocation of this Technical Specification to the TRM does not degrade the performance of any safety systems or prevent actions assumed in the accident analysis, nor does it alter any of the assumptions made in the analysis that could increase the consequences of accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Revision of Index Pages IX and XIII and the proposed change to Bases sections, by relocating them to the TRM, are administrative changes. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated. The proposed changes do not alter how any structure, system, or component functions. There will be no effect on equipment important to safety. The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated, nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. 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 relocation of Technical Specification 3/4.9.3.2, ''Refueling Operations, Spent Fuel Pool Temperature,'' to the TRM does not imply any reduction in its importance in limiting the spent fuel pool bulk temperature to less than or equal to 140 °F. The proposed change will not alter the way pool temperature is measured. It will not alter any of the assumptions used in the spent fuel pool fuel handling accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. The proposed relocation of Technical Specification 3/4.9.3.3, ''Refueling Operations, Decay Time,'' to the TRM does not imply any reduction in its importance in insuring that the most recent core offload has decayed a sufficient time. The proposed change will not alter the requirement that the most recent core offload has decayed a sufficient time, it will not alter any of the assumptions used in the spent fuel pool fuel handling accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. The relocation of Technical Specification 3/4.9.5, ''Refueling Operations, Communications,'' to the TRM does not imply any reduction in its importance in insuring communication between the control room and the refueling station. The proposed change will not alter the requirement on communication between the control room and the refueling station, it will not alter any of the assumptions used in the spent fuel pool fuel handling accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. The relocation of Technical Specification 3/4.9.6, ''Refueling Operations, Crane Operability—Containment Building,'' to the TRM does not imply any reduction in its importance in insuring that the lifting device on the refueling machine has adequate capacity. The proposed change will not alter the requirement that the lifting device on the refueling machine has adequate capacity, it will not alter any of the assumptions used in the accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. The relocation of Technical Specification 3/4.9.7, ''Refueling Operations, Crane Travel—Spent Fuel Storage Pool Building,'' to the TRM does not imply any reduction in its importance in insuring that loads in excess of 1800 pounds (except of a consolidated fuel storage box) are prohibited from travel over irradiated fuel. The proposed change will not alter the requirement that the crane interlocks and/or physical stops are OPERABLE, it will not alter any of the assumptions used in the spent fuel pool fuel handling accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. Revision of Index Pages IX and XIII and the proposed change to Bases sections by eliminating the sections corresponding to the relocated Technical Specifications are administrative changes. These changes will not alter any of the assumptions used in the spent fuel pool fuel handling accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed changes will not result in a significant reduction in a margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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.

PECO Energy Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of amendment request: June 7, 1999.

Description of amendment request: The proposed change to the Technical Specifications (TSs), if approved, will reflect the permanent deactivated configuration of the ‘‘wet’’ instrument reference leg isolation valve HV–61–102 which originally connected the Drywell Floor and Equipment Drain Tanks to level instruments outside the containment. The TS changes affecting TS Table 3.6.3–1, ‘‘Primary Containment Isolation Valves,’’ and its associated notations will reflect the current plant configuration. More specifically, TS Section 3/4.6.3, ‘‘Primary Containment Isolation Valves,’’ Table 3.6.3–1, Penetration Number 230B will be revised to designate the function of valve HV–61–102 as ‘‘Deactivated,’’ the maximum isolation time for valve HV–61–102 will be eliminated, and notations 1, 23, and 29 will be replaced with a new notation indicating the permanent configuration of the subject valve.

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

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

The closed valve, HV–61–102, has no effect on the function of the Drywell Sump/ Equipment Drain Tanks, other safety-related systems, or other containment penetrations. The current status of the valve is locked closed, de-energized, and the motor operator cannot be accidentally actuated. In addition, the line is capped downstream of the isolation valve. As described above, the valve is considered to be in a passive configuration, where a malfunction is not expected and cannot cause an increase in the probability of a malfunction to itself or other safety-related equipment. The potential for increased releases outside the containment due to breaching of the valve assembly is no greater than that of the isolation design previously evaluated.

Therefore, the proposed change to the TSs does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated in the Safety Analysis Report.

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

The abandoned isolation valve conforms to approved isolation configurations, and its structural integrity has not been degraded by the modified configuration. The original function of valve HV–61–102 was only to provide isolation of the instrument line. Following the modification, the valve is independent of the function of the Drywell Sump/ Equipment Drain Tanks, other safety-related systems, and other penetrations. Since the valve is passive and has no requirements to be operated, it cannot create a different type of malfunction on itself or other safety-related systems. In addition, the valve is specifically designed to isolate and is essentially passive during accident conditions, it has no activity that could be the initiator of an accident of a different type.

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

3. The proposed TS changes do not involve a significant reduction in a margin of safety. Isolation valve HV–61–102 in its proposed permanent configuration meets the margin of safety described in TS Bases 3/4.6.3 since it is kept closed under all operational conditions and will not be under the constraint of TS closing times in order to maintain releases within specifications. The proposed changes have no impact on any safety analysis assumptions.

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

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

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: July 23, 1999, as supplemented on September 13, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement 4.8.1.1.2 to allow the 24-hour emergency diesel generator endurance run to be performed during power operation (i.e., Modes 1 and 2) instead of restricting the test to when the reactor was shutdown.

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

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

The proposed change to Technical Specification Surveillance Requirement (SR) 4.8.1.1.2.d.7 (24-hour emergency diesel generator (EDG) endurance run test) to eliminate the restriction to perform the test during shutdown conditions does not involve a significant increase in the probability of any previously evaluated accident. Although paralleling or connecting the EDG to off-site power for the test could induce an electrical distribution system perturbation, the same possibility exists when the EDG is tested during the monthly 1-hour loaded surveillance test (SR 4.8.1.1.2 a 2). This risk during testing the EDG monthly at power was reviewed and found acceptable by the NRC. Further, none of the automatic actuations and interlocks in the tested portion of the electrical system or the EDG control system are disabled during the 24-hour endurance run. Thus, the onsite safety-related electrical system remains protected from potential faults and perturbations.

The ability and capability [o]f the EDG to perform their safety function (mitigate the consequences of a previously evaluated accident) is also unaffected. This capability was demonstrated not only by the tests conducted in the EDG manufacturer's plant, but continue to be demonstrated by surveillance testing performed at the station.

This testing verifies specific design criteria, which assure continued EDG operability even during testing. Examples of presently performed Technical Specification testing that demonstrate the ability and capability of the EDG to perform its safety functions are:

• SR 4.8.1.1.2. d. 2 requires, in part, that on a load rejection of greater than 820 KW, the voltage and frequency be restored to acceptable values within 4 seconds.

• This surveillance demonstrates the ability of the EDGs to withstand a loss of load, as it would occur in a normal safeguards equipment controller (SEC) actuation, without compromising its ability to be ready to accept a new loading sequence and carry its design safety function.

• SR 4.8.1.1.2. d. 9 requires, in part, that with the EDG operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power.

This surveillance demonstrates the ability of the EDGs to be disconnected from the grid, if in a test mode, on an accident signal, and be ready to accept a new loading sequence and carry its design safety function.

• SR 4.8.1.1.2. a. 2 requires, in part, that every 31 days each EDG be demonstrated OPERABLE by synchronizing it to the grid for greater than or equal to 60 minutes.

Note that this proposed amendment request eliminates a discrepancy between the current requirement to perform the 24 hour run during shutdown and SR 4.8.1.1.2.a.2, which would allow a 24 hour run at power.

Additionally, PSE&G performed an assessment of the potentially added risk of an additional 24 hours of on-line EDG testing. The unavailability of all three EDGs was increased in the Probabilistic Safety Analyses (PSA) for both Salem Units 1 and 2 to correspond to an additional 24 hours per cycle out-of-service time each 18-month operating cycle. The unavailability was changed from 1.86E–02/year to 2.0E–2/year. The increase in the baseline internal events core damage frequency (CDF) was determined to be 1.6E–07 events/year for both Salem Units 1 and 2. Based on the definition provided in Regulatory Guide 1.174, Paragraph 2.2.4, this increase is considered a very small increase in risk (less than 1.0E–06 events/year).

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

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

The proposed amendment to Technical Specification Surveillance Requirement 4.8.1.1.2.d.7 (24-hour endurance run test) to eliminate the restriction to perform the test during shutdown conditions does not physically modify the facility, introduce a new failure mode, or propose a different operational mode of the AC electrical power sources, or Emergency Diesel Generators.

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

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

The AC Electrical distribution system has been designed to provide sufficient redundancy and reliability to ensure the availability of the EDGs to provide the required safety function under design basis events to protect the power plant, the public and plant personnel. Specifically, the ability of the EDGs to separate from the off-site power source has been designed and tested per Technical Specifications requirements.

Performance of the 24-hour endurance run during power operations will not affect the availability of any of the required power sources, nor the capability of the EDGs to perform their intended safety function. Furthermore, performing the test when the undervoltage protection of the 4160–V vital buses required by the Salem Station Technical Specification 3.3.2.1 is operable, provides for an added level of protection to the EDG that is not available while 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.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

NRC Section Chief: James W. Clifford.

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

Date of application for amendments: August 30, 1999 (TS 99–08).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Technical Specification (TS) requirements to provide alternatives to the requirement of actually measuring response times.

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

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

This change to the TS does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS [Reactor Trip System] and engineered safety feature actuation system (ESFAS) instrumentation is being used, the time response allocations/modeling assumptions in the [Final Safety Analysis Report] Chapter 15 analyses are still the same, only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the Final Safety Analysis Report. 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.

This change does not alter the performance of pressure [or] differential pressure transmitters, solid state protection system racks, nuclear instrumentation, or input and output master/slave relays used in the plant protection systems. Applicable sensors, solid state protection system (SSPS) racks, nuclear instrumentation, and relays will still have response time verified by test prior to placing the equipment in operational service and after any maintenance that could affect the response time of that equipment. Changing the method of periodically verifying instrument response time for certain instruments from RTT [Response Time Test] to calibration and channel checks or functional test will not create any new accident initiators or scenarios. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and pressure differential sensors and SSPS racks, nuclear instrumentation, or logic systems is modified to allow use of actual test data or engineering data (various Westinghouse WCAPs [topical reports]). The method of verification still provides assurance that the total system response time is within that assumed in the safety analysis, since calibration checks and functional tests will detect any degradation which might significantly affect equipment response time. Therefore, the proposed license amendment request does not result in a significant reduction in margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Section Chief: Sheri R. Peterson.

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

Date of application for amendments: August 30, 1999 (TS 99–10).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Technical Specifications (TS) to provide clarification to the requirements for containment isolation valves.

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

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

The proposed revisions enhance the technical specification (TS) requirements to provide greater consistency with the standard TS in NUREG–1431. This revision proposes changes to the requirements for containment isolation valves in Specifications 3.6.3. A proposed revision relocates a surveillance requirement (SR) from SQN TS 3.6.1.1, ''Containment Integrity'' to SQN TS 3.6.3, ''Containment Isolation Valves.'' A proposed revision to TS 3.6.3, Action (a), a new Action (b), and a proposed revision to SR 4.6.3.2 provide improvements to the existing TS requirements. The proposed revisions are not the result of changes to plant equipment, system design, testing methods, or operating practices. The modified requirements will allow some relaxation of current action requirements, and SRs. These changes provide more appropriate requirements in consideration of the safety significance and the design capabilities of the plant as determined by the improved standard TS industry effort. SQN TS 3.6.3, ''Containment Isolation Valves,'' continues to provide controls to ensure these valves isolate within the time limits assumed in the safety analyses. Operability of these valves continues to assure that the containment isolation function assumed in the safety analyses is maintained. Since these proposed revisions will continue to support the required safety functions without modification of the plant features, the probability of an accident is not increased.

The provisions proposed in this change request will continue to maintain an acceptable level of protection for the health and safety of the public and will not significantly impact the potential for the offsite release of radioactive products. The overall effect of the proposed change will result in specifications that have equivalent or improved requirements compared to existing specifications for containment isolation valve operability and will not significantly increase the consequences of an accident.

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

The proposed revisions are not the result of changes to plant equipment, system design, testing methods, or operating practices. The modified requirements will allow some relaxation of current action requirements, and a SR consistent with NUREG–1431. These changes provide more appropriate requirements in consideration of the safety significance and the design capabilities of SQN's containment isolation system. The specifications for containment isolation valves serve to provide controls for maintaining the containment pressure boundary. TVA's proposed changes does not contribute to the generation of postulated accidents. Since the function of the containment isolation valves and their associated systems remains unchanged, and the effects do not contribute to accident generation, the proposed changes will not create the possibility of a new or different kind of accident.

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

The proposed changes will not result in changes to system design or setpoints that are intended to ensure timely identification of plant conditions that could be precursors to accidents or potential degradation of accident mitigation systems. Operability requirements for SQN's containment isolation valves remain unchanged. TVA's proposed revisions provide some relaxation and flexibility to existing actions and a SR; however, the addition of a new action requirement for a 31-day periodic verification of valve position provides conservative administrative controls to ensure containment isolation function is maintained. The action times are acceptable considering the redundant features of containment penetration flow paths and the allowed time intervals that have been developed by the industry and NRC.

TVA's revisions will continue to provide the necessary actions to minimize the impact of inoperable containment isolation valves and will provide testing activities that will ensure containment isolation system operability. The setpoints and design features that support the margin of safety are unchanged and actions for inoperable systems continue to provide appropriate time limits and compensatory measures. Accordingly, the proposed changes will not significantly reduce the margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Section Chief: Sheri R. Peterson.

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

Date of application for amendments: August 30, 1999 (TS 99–11).

Brief description of amendments: The proposed amendments would add Sequoyah (SQN) Technical Specification (TS) 3.0.7 to address the use of interim provisions upon discovery of unintended TS action.

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

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

TVA proposes the addition of a new definition and limiting condition for operation (LCO) that will allow the interim correction of erroneous TS requirements until NRC's review of an amendment request is completed. This allowance will only apply to those errors that are clearly in conflict with the intended purpose of the TS requirement. The proposed revision will not alter any plant equipment or operating practices or deviate from the intended application of the TS requirements. Therefore, the probability of an accident is not increased by this revision. Likewise, the consequences of an accident is not increased because the proposed allowance will maintain the underlying intent of the TS requirements, the plant licensing basis, and plant nuclear safety.

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

The proposed revision to the SQN TSs will not alter plant equipment or operating practices. The intent of the TS requirements will be maintained to ensure the assumed initial conditions for accidents and the availability of mitigation systems in the event of an postulated accident. The proposed addition will not promote activities that have the potential to generate accidents. Therefore, the proposed revision will not create the possibility of an accident of a new or different kind.

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

TVA's proposed revision to add an allowance to correct erroneous TS requirements will not alter plant systems or those setpoints and limits that are use[d] to maintain safety functions. Any corrections implemented in accordance with the proposed allowance will be consistent with the underlying intent of the TSs. TVA will pursue timely correction of such errors through the license amendment process while temporarily utilizing the corrected requirement. This will ensure that inadequate TS requirements are resolved with NRC in an acceptable time interval. Implementation of the proposed revision will enhance the ability to maintain the licensing basis and safety features of the plant without the need for unnecessary unit shutdowns or regulatory activities. Therefore, the proposed revision maintains the plant safety features without the reduction of any margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Section Chief: Sheri R. Peterson

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

Date of application request: September 8, 1999.

Description of amendment request: The amendment will authorize revisions to the Final Safety Analysis Report (FSAR) to reflect increases in the radiological dose consequences in the Callaway FSAR for the steam generator tube rupture (SGTR) and main steam line break (MSLB) accidents.

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

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

This change increases the offsite dose consequences for the MSLB and SGTR accidents reported in FSAR Sections 15.1 and 15.6. Non-conservative assumptions regarding letdown flow rate, iodine isotopic mix in the source term, resin effeciency, and termination of the flash release pathway were identified in the SGTR and MSLB radiological consequence analyses. The correction of these non-conservative assumptions results in an increase in the radiological consequences reported in FSAR Tables 15.1–4 and 15.6–5. However, these increases are not significant since the new values remain less than the 10 CFR 100.11 regulatory requirements and the guideline values provided by the Standard Review Plan [NUREG–0800].

There will be no increase in the probability of previously evaluated accidents. This change only involves the modeling and calculation of the SGTR and MSLB radiological consequences. [There are no equipment or system changes.] Protection system performance will remain within the assumptions of the previously performed accident analyses since no hardware changes are proposed. The protection systems will continue to function in a manner consistent with the plant design basis. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of, nor an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

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

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

This change is the result of a re-analysis of the MSLB and SGTR radiological consequences. These accidents were previously analyzed in the FSAR. None of the changes in the dose calculation modeling create the possibility of a new or different kind of accident.

There are no hardware changes associated with this amendment application nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not affect the normal method of plant operation, other than the imposition of administrative limits on the concentrations of I–134 [Iodine-134] and Dose Equivalent I–131 until this amendment application is approved by NRC. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change.

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

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

The re-analysis of the MSLB and SGTR radiological consequences, and the resultant increase in consequences reported in FSAR Tables 15.1–4 and 15.6–5, ensures that the accident analyses support the plant operating conditions allowed by current Technical Specification 3.4.8, Reactor Coolant System Specific Activity (ITS [Improved Technical Specification] 3.4.16), and current Technical Specification 3.7.1.4, Plant Systems Specific Activity (ITS 3.7.18).

The proposed change does not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR [departure from nucleate boiling ratio], F_Q [heat flux hot channel factor], FdeltaH [nuclear enthalpy rise hot channel factor], LOCA PCT [peak cladding temperature for the loss-of-coolant accident], peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan continue to be met.

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

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

Local Public Document Room location: Elmer Ellis Library, University of Missouri, Columbia Missouri 65201.

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

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

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

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

Date of amendment requests: September 14, 1998.

Description of amendment requests: The proposed amendments would change the runout limits for a safety injection (SI) pump to 675 gallons per minute (gpm), unless the pump is specifically tested to a higher flow rate, not exceeding 700 gpm for both Units 1 and 2. This change was initiated upon reevaluation of correspondence from Westinghouse sent to the licensee in 1991, which indicated that the generic runout limits for Pacific 2'' JTCH pumps was 675 gpm unless each specific pump is tested to a higher flow rate. Individual testing is necessary due to test variations between pumps which may limit the applicability of testing of one pump to another pump due to manufacturing tolerances in the sand cast impellers and material changes in the pump casing.

Furthermore, the bases section is being clarified to describe why the injection rather than the recirculation mode during flow balancing is the minimum resistance and, consequently, more conservative configuration for runout considerations.

Date of publication of individual notice in **Federal Register**: August 31, 1999 (64 FR 47533).

Expiration date of individual notice: September 30, 1999

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

Date of publication of individual notice in **Federal Register**: August 31, 1999 (64 FR 47533).

Expiration date of individual notice: September 30, 1999.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

Date of application for amendments: October 8, 1998.

Brief description of amendments: The amendments would revise Technical Specification (TS) 3.3.3.8 for Unit 1 and TS 3.3.3.6 for Unit 2, ''Post-Accident Instrumentation.'' The proposed changes to the TSs will place tighter restrictions on the amount of time the refueling water storage tank (RWST) water level instrumentation may be inoperable before the limiting conditions for operation in the TSs are applied.

Date of publication of individual notice in **Federal Register**: August 31, 1999 (64 FR 47532).

Expiration date of individual notice: September 30, 1999.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

Date of application for amendments: December 3, 1998.

Brief description of amendments: The amendments would make administrative changes to several Technical Specifications to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors.

Date of publication of individual notice in **Federal Register**: August 31, 1999 (64 FR 47535).

Expiration date of individual notice: September 30, 1999.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

Date of application for amendments: May 21, 1999.

Brief description of amendments: The amendments would change the Technical Specifications (T/S) to allow reactor coolant system temperature changes in certain Mode 5 and 6 action statements if the shutdown margin is sufficient to accommodate the expected temperature change. In addition, footnotes regarding additions of water from the refueling water storage tank to the reactor coolant system are clarified and relocated to action statements. Additional actions are added in Table 3.3–1, ''Reactor Trip System Instrumentation,'' when the required source range neutron flux channel is inoperable. Corresponding changes are proposed for the bases for T/S 3/4.1.1, ''Boration Control,'' and T/S 3/4.1.2, ''Boration Systems.'' Administrative changes are proposed to improve clarity. Finally, additions are made to shutdown margin T/S surveillance requirements to address use of a boron penalty (requirement for additional boron) during residual heat removal system operation in Modes 4 and 5.

Date of publication of individual notice in **Federal Register**: July 12, 1999 (64 FR 37574).

Expiration date of individual notice: August 11, 1999.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: September 23, 1998, as supplemented on December 7, 1998, and August 10, 1999.

Brief description of amendment: This amendment revises Technical Specification (TS) 3/4.6.1.3, ''Containment Air Locks,'' and its associated bases, to clarify the requirements for locking an air lock door shut and to make it consistent with NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995.

Date of issuance: September 14, 1999.

Effective date: September 14, 1999.

Amendment No.: 90.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications

Date of initial notice in Federal Register: October 21, 1998 (63 FR 56239)

The December 7, 1998, and August 10, 1999, submittals contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: June 15, 1999.

Brief description of amendment: This amendment changes the Technical Specifications to incorporate the performance-based 10 CFR 50 Appendix J, Option B for Type A tests (containment integrated leakage rate tests). Option B will be implemented for Type A testing in accordance with NRC Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' dated September 1995, and Nuclear Energy Institute (NEI) Guideline 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' dated July 26, 1995. Type B and C testing (containment penetration leakage tests) will continue to be performed in accordance with 10 CFR 50 Appendix J, Option A.

Date of issuance: September 17, 1999.

Effective date: September 17, 1999.

Amendment No.: 91.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 14, 1999 (64 FR 38023). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

Date of application for amendments: July 30, 1999.

Brief description of amendments: The amendments changed the maximum allowable temperature of the ultimate heat sink in the technical specifications from 98 degrees Fahrenheit to 100 degrees Fahrenheit. The change is in effect from the date of this amendment until September 30, 1999.

Date of issuance: September 8, 1999.

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

Amendment Nos.: 103 and 103.

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

Public comments requested as to proposed no significant hazards consideration: Yes (64 FR 44962 dated August 18, 1999). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by September 17, 1999, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendments. The Commission's related evaluation of the amendments, finding of exigent circumstances and final no significant hazards consideration determination are contained in a Safety Evaluation dated September 8, 1999.

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of application for amendments: May 3, 1999, as supplemented by letter dated September 10, 1999.

Brief description of amendments: The amendments relocated the requirements of Technical Specification (TS) Section 3/4.6.I to the Updated Final Safety Analysis Report (UFSAR). TS Section 3/4.6.I contains reactor coolant chemistry limiting conditions for operation (LCO) and surveillance requirements (SR) for conductivity, chloride concentration, and pH.

Date of issuance: September 23, 1999.

Effective date: Immediately, to be implemented within 30 days including relocation of the removed TSs and associated bases to the licensee's UFSAR pending change file. In addition, the licensee shall include the relocated information in the UFSAR submitted to the NRC, pursuant to 10 CFR 50.71(e), except for any information that has been changed in accordance with 10 CFR 50.59 and described in the change summaries submitted to NRC pursuant to 10 CFR 50.59.

Amendment Nos.: 173 & 169.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 11, 1999 (64 FR 43768). The September 10, 1999, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

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

Date of application for amendments: June 29, 1999.

Brief description of amendments: The amendments increased the notch testing surveillance interval of partially withdrawn control rods in Technical Specification Surveillance Requirement 3/4.3.C, ''Reactivity Control—Control Rod Operability,'' from an interval of once in 7 days to once in 31 days.

Date of issuance: September 23, 1999.

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 190 & 187.

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

Date of initial notice in Federal Register: July 28, 1999 (64 FR 40905).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: May 24, 1999

Brief description of amendments: The amendments revise the maximum local fuel pin centerline temperature safety limit in Technical Specification 2.1.1.1 from the limit determined using the TACO2 fuel performance computer code to the value determined using a newer TACO3 computer code.

Date of Issuance: September 24, 1999.

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

Amendment Nos.: Unit 1—306, Unit 2—306, Unit 3—306.

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

Date of initial notice in Federal Register: June 30, 1999 (64 FR 35203).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

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

Date of application for amendments: July 22, 1998, and supplemented by letters dated October 22, 1998, January 28, May 6, June 24, August 17 and September 15, 1999.

Brief description of amendments: The amendments revise various sections of the Technical Specifications (Appendix A of the Catawba operating licenses) to permit use of Westinghouse's Robust Fuel Assemblies for future core reloads.

Date of issuance: September 22, 1999.

Effective date: As of the date of issuance and shall be implemented prior to beginning the installation of the Westinghouse fuel, currently projected to be Fuel Cycle 13 and 11 for Units 1 and 2, respectively.

Amendment Nos.: Unit 1—180; Unit 2—172.

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

Date of initial notice in Federal Register: November 18, 1998 (63 FR 64108); May 19, 1999 (64 FR 27317); August 11, 1999 (64 FR 43770) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina.

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

Date of application for amendments: July 22, 1998, and supplemented by letters dated October 22, 1998, and January 28, May 6, June 24, August 17 and September 15, 1999

Brief description of amendments: The amendments revise various sections of the Technical Specifications (Appendix A of the McGuire operating licenses) to permit use of Westinghouse's Robust Fuel Assemblies for future core reloads.

Date of issuance: September 22, 1999.

Effective date: As of the date of issuance and shall be implemented prior to beginning the installation of the Westinghouse fuel, currently projected to be Fuel Cycle 15 and 14 for Units 1 and 2, respectively.

Amendment Nos.: Unit 1—188; Unit 2—169.

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

Date of initial notice in Federal Register: August 11, 1999 (64 FR 43771); June 30, 1999 (64 FR 35202); December 16, 1998 (64 FR 69388)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina

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

Date of amendment request: April 9, 1999, as supplemented by letter dated July 29, 1999

Brief description of amendment: The amendment revises the requirements associated with the station batteries and the direct current (DC) sources to the 125 volt DC switchyard distribution system.

Date of issuance: September 14, 1999.

Effective date: As of the date of issuance and shall be implemented within 45 days from the date of issuance (including issuance of the Technical Requirements Manual for use by licensee personnel).

Amendment No.: 200.

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

Date of initial notice in Federal Register: May 19, 1999 (64 FR 27321).

The July 29, 1999, letter provided clarifying and additional information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

Date of amendment request: June 1, 1999, as supplemented by letters dated July 29 and August 19, 1999.

Brief description of amendment: The amendment revised the Technical Specifications to allow, under specific conditions, certain once-through steam generator (OTSG) tubes with tube end crack indications adjacent to the primary cladding region of the upper and lower OTSG tubesheets to remain in service.

Date of issuance: September 14, 1999.

Effective date: As of the date of issuance and shall be implemented prior to reactor startup after refueling outage 1R15.

Amendment No.: 201.

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

Date of initial notice in Federal Register: June 30, 1999 (64 FR 35205).

The July 29 and August 19, 1999, letters provided clarifying information that did not change the scope of the June 1, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

Date of application for amendment: May 17, 1999.

Brief description of amendment: The amendment changes Technical Specification Section 3.3.8, ''Emergency Diesel Generator Loss of Power Start,'' Surveillance Requirement 3.3.8.1 and corresponding basis section. The surveillance is revised to make a note included in the surveillance consistent with the method of performing the surveillance.

Date of issuance: September 13, 1999.

Effective date: September 13, 1999.

Amendment No.: 187.

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

Date of initial notice in Federal Register: July 14, 1999 (64 FR 38026).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: December 23, 1998.

Brief description of amendment: The proposed amendment revised the surveillance frequency for verifying the operability of motor-operated isolation valves and condensate makeup valves in the Isolation Condenser Technical Specification 4.8.A.1 and Bases page from once per month to once per 3 months.

Date of Issuance: September 24, 1999.

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

Amendment No.: 209.

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

Date of initial notice in Federal Register: April 7, 1999 (64 FR 17026).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: October 19, 1998, as supplemented August 19, 1999.

Brief description of amendment: The proposed amendment adds operability and surveillance requirements to the Technical Specifications for the remote shutdown system similar to the standard technical specifications for Babcock & Wilcox nuclear plants as described in NUREG–1430.

Date of issuance: September 22, 1999.

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

Amendment No.: 216.

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

Date of initial notice in Federal Register: November 18, 1998 (63 FR 64118). The August 19, 1999, supplement to the application did not change the staff's proposed no significant hazards consideration determination or expand the scope of the application as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (Regional Depository) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: April 30, 1999.

Brief description of amendment: The amendment revises Duane Arnold Energy Center (DAEC) Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to revise the safety function lift setpoint tolerance limits for the main safety valves (SVs) and the safety/relief valves (SRVs).

Date of issuance: September 22, 1999.

Effective date: September 22, 1999, to be implemented within 30 days.

Amendment No.: 228.

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

Date of initial notice in Federal Register: July 14, 1999 (64 FR 38028).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, IA 52401.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: May 15, 1998, as supplemented by letters dated September 25, October 13, December 9 (two letters), 1998; January 11, April 1, and April 22, 1999.

Brief description of amendment: This amendment changes Technical Specification (TS) 5.5, ''Storage of Unirradiated and Spent Fuel,'' to reflect a planned modification to increase the storage capacity of the spent fuel pool from 2776 to 4086 fuel assemblies. It also deletes an inappropriate statement and reference within TS 5.5.

Date of issuance: June 17, 1999.

Effective date: This license amendment is effective as of the date of its issuance to be implemented before spent fuel is stored within the new high-density spent fuel rack modules authorized for installation and use by this amendment.

Amendment No.: 167.

Facility Operating License No. DPR–63: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: November 24, 1998 (63 FR 64973).

The September 25, October 13, December 9 (two letters) 1998, January 11, April 1, and April 22, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents

Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

Date of amendment request: June 23, 1999.

Description of amendment request: To revise Technical Specification (TS) 3.7.6.2 to increase the allowable outage time for the Control Room Air Conditioning Subsystem from 30 days to 60 days, on a one-time basis for each train, to allow adequate time to replace portions of the existing system during the current operating cycle, and to exclude the requirements of TS 3.0.4 and TS 4.0.4 during the implementation of the modification.

Date of issuance: September 17, 1999.

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

Amendment No.: 62.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications/License.

Date of initial notice in Federal Register: July 14, 1999 (64 FR 38032).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of application for amendment: March 17, 1999.

Brief description of amendment: The amendment changes Technical Specifications 3.5.2, ''Emergency Core Cooling Systems—ECCS Subsystems—Tavg ≥300 °F;'' 3.7.1.7, ''Plant Systems—Atmospheric Steam Dump Valves;'' and 3.7.6.1, ''Plant Systems—Control Room Emergency Ventilation System.'' The changes will revise: (1) Surveillance requirements for the Emergency Core Cooling System valves, (2) the atmospheric steam dump valve requirements to focus on the steam release path instead of the individual valves, and (3) the allowed outage time for the atmospheric steam valves and Control Room Emergency Ventilation System. The licensee made changes to the Bases pages consistent with the proposed changes to the TSs.

Date of issuance: August 12, 1999.

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

Amendment No.: 238.

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

Date of initial notice in Federal Register: April 21, 1999 (64 FR 19559).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

Date of application for amendment: June 4, 1999.

Brief description of amendment: The amendment makes administrative changes to the Technical Specifications.

Date of issuance: September 14, 1999.

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

Amendment No.: 193.

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

Date of initial notice in Federal Register: July 28, 1999 (64 FR 40906).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendment: January 28, 1999, as supplemented April 29, 1999, and May 17, 1999. By letters dated April 29, 1999, and May 17, 1999, the licensee revised the original submittal dated January 28, 1999, in response to questions raised by the NRC staff.

Brief description of amendment: The amendment changes the Technical Specifications by reducing the number of emergency diesel generators required to be operable under certain conditions.

Date of issuance: September 14, 1999.

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

Amendment No.: 194.

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

Date of initial notice in Federal Register: June 2, 1999 (64 FR 29713). This notice superceded a notice dated April 21, 1999 (64 FR 19563).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendment: January 28, 1999, as supplemented July 16, 1999.

Brief description of amendment: The amendment removes lists of containment isolation valves from the Technical Specifications (TSs) and modifies the TSs accordingly.

Date of issuance: September 16, 1999.

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

Amendment No.: 195.

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

Date of initial notice in Federal Register: May 19, 1999 (64 FR 27323).

The July 16, 1999, submittal did not change the staff's initial proposed finding of no significant hazards considerations.

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 16, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendment: April 5, 1999.

Brief description of amendment: The proposed changes would revise Appendix A (Section 6.1) and Appendix B (Section 7.1) of the James A. FitzPatrick Technical Specifications. The proposed changes would remove the position title of General Manager from these sections and would state that if the Site Executive Officer is unavailable, he will delegate his responsibilities to another staff member, in writing. In addition the position title of Resident Manager, used in Appendix B, Section 7.1, would be replaced by Site Executive Officer.

Date of issuance: September 13, 1999.

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

Amendment No.: 254.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications and the Environmental Technical Specifications.

Date of initial notice in Federal Register: August 11, 1999 (64 FR 43775).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

Date of application for amendment: October 8, 1997.

Brief description of amendment: The amendment revises actions in the Technical Specifications to be taken in the event multiple control rods are inoperable.

Date of issuance: September 21, 1999.

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

Amendment No.: 255.

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

Date of initial notice in Federal Register: February 11, 1998 (63 FR 6991).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 21, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: December 30, 1998, as supplemented September 13, 1999.

Brief description of amendment: This amendment revises Technical Specification (TS) Limiting Condition for Operation 3.7.3 and TS Table 3.7.3–1. These changes modify the flood protection actions required when severe storm warnings that may affect the site are in effect or during periods of elevated river water level.

Date of issuance: September 17, 1999.

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

Amendment No.: 122.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in Federal Register: February 24, 1999 (64 FR 9200).

The September 13, 1999, supplement provided clarifying information that did not change the initial proposed no significant hazards determination or expand the scope of the initial **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: May 24, 1999, as supplemented June 21, 1999.

Brief description of amendment: This amendment revises the Technical Specifications (TSs) to correct typographical and editorial errors, and is considered administrative in nature.

Date of issuance: September 21, 1999

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

Amendment No.: 123.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in Federal Register: June 30, 1999 (64 FR 35209).

The June 21, 1999, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 21, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070

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

Date of application for amendments: July 2, 1999.

Brief description of amendments: The amendments delete TS 3/4.3.4, ''Instrumentation—Turbine Overspeed Protection,'' and its associated Bases and relocate the requirements to the licensee-controlled Updated Final Safety Analysis Report.

Date of issuance: September 14, 1999

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

Amendment Nos.: 224 and 205.

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

Date of initial notice in Federal Register: August 11, 1999 (64 FR 43776).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 14, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

Date of application for amendments: December 31, 1998 (PCN–501), as supplemented June 14, 1999.

Brief description of amendments: The amendments consist of changes to Technical Specification 3.3.5, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' and will include restrictions on operation with a channel of the refueling water storage tank level—low input to the recirculation actuation signal and the steam generator pressure—low input or steam generator pressure difference—high input to the emergency feedwater actuation signal in the tripped condition.

Date of issuance: September 7, 1999.

Effective date: September 7, 1999, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—157; Unit 3—148.

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

Date of initial notice in Federal Register: July 28, 1999 (64 FR 40907).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 7, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

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

Date of application for amendments: June 18, 1997 (PCN–478), as supplemented May 24 and August 10, 1999.

Brief description of amendments: The amendments modify the Technical Specification surveillance requirements related to diesel generator testing to more clearly reflect safety analysis and testing conditions as it is performed.

Date of issuance: September 9, 1999.

Effective date: September 9, 1999, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—158; Unit 3—149.

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

Date of initial notice in Federal Register: December 31, 1997 (62 FR 68315) The licensee's letters dated May 24 and August 10, 1999, provided updated Technical Specification pages, clarifications, and additional information that were within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 9, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room Location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: June 7, 1999.

Brief description of amendments: The amendments revised Technical Specification (TS) 2.2.1, ''Reactor Trip System (RTS) Instrumentation Setpoints,'' and TS 3.3.2, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' and the associated Bases, by removing the Total Allowance, Sensor Error, and Z terms (Z is the statistical summation of errors excluding sensor and rack drift) from the RTS and ESFAS Instrumentation Trip Setpoints Tables. This replaces the five-column methodology with a two-column methodology that consists of the trip setpoint and allowable value columns.

Date of issuance: September 13, 1999.

Effective date: September 13, 1999, to be implemented within 30 days.

Amendment Nos.: Unit 1—116; Unit 2—104.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 30, 1999 (64 FR 35211) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

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

Date of application for amendments: June 24, 1999 (TS 99–06).

Brief description of amendments: The amendments revise the Sequoyah Nuclear Plant Technical Specifications (TS) by adding a footnote to allow use of an installed spare electrical inverter, if needed.

Date of issuance: September 23, 1999.

Effective date: As of the date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: 246 and 237.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TS.

Date of initial notice in Federal Register: August 2, 1999 (64 FR 41973) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

Date of amendment request: June 23, 1999, as supplemented by letter dated August 4, 1999.

Brief description of amendments: The amendments revise Surveillance Requirement 3.8.1.13, ''AC Sources—Operating'' to clarify that each emergency diesel generator automatic noncritical trip, except for engine overspeed and generator differential current, is bypassed on either a loss-of-offsite power or a safety injection actuation signal.

Date of issuance: September 21, 1999.

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

Amendment Nos.: 69 and 69.

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

Date of initial notice in Federal Register: July 14, 1999 (64 FR 38037) The August 4, 1999, letter provided additional and clarifying information that did not change the scope of the June 23, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 21, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

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

Date of amendment request: February 12, 1999, as supplemented by letter dated June 14, 1999

Brief description of amendments: The amendments change Technical Specification (TS) 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,'' TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and TS 5.6.10, ''Steam Generator Tube Inspection Report,'' to implement the 1.0 Volt Steam Generator Tube Repair Criteria for CPSES, Unit 1.

Date of issuance: September 22, 1999.

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

Amendment Nos.: Unit 1—Amendment No. 70; Amendment No. 70.

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

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24202) The June 14, 1999, supplement provided clarifying information that did not change the scope of the February 12, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

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

Date of amendment request: October 2, 1998, as supplemented by letters dated July 27 and August 26, 1999.

Brief description of amendments: The amendments revise Technical Specfications for CPSES, Unit 1, to define the F* steam generator tube plugging criteria in TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and associated reporting requirements in TS 5.6.10, ''Steam Generator Inspection Report.''

Date of issuance: September 22, 1999.

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

Amendment Nos.: Unit 1—Amendment No. 71; Unit 2—Amendment No. 71.

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

Date of initial notice in Federal Register: November 4, 1998 (63 FR 59597). The July 27 and August 26, 1999, letters provided clarifying information that did not change the scope of the October 2, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

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

Date of application for amendment: May 5, 1999.

Brief description of amendment: The amendment revises the technical specifications (TSs) to enhance the limiting conditions for operation and surveillance requirements relating to the standby liquid control system and to incorporate certain provisions of NRC's rule on anticipated transients without scram. The change involves the use of enriched boron in the standby liquid control system and improves upon other aspects of the TSs for this system.

Date of Issuance: September 17, 1999.

Effective date: September 17, 1999, and shall be implemented within 30 days.

Amendment No.: 175.

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

Date of initial notice in Federal Register: June 30, 1999 (64 FR 35214).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

Date of application for amendment: July 12, 1999.

Brief description of amendment: The amendment revises the values for the minimum critical power ratio safety limits and deletes the wording classifying the limits as cycle-specific values.

Date of Issuance: September 21, 1999.

Effective date: September 21, 1999, and shall be implemented within 60 days.

Amendment No.: 176

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

Date of initial notice in Federal Register: July 28, 1999 (64 FR 40910).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 21, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

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

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

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 5, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

Duke Energy Corporation, Docket No. 50–369, McGuire Nuclear Station, Unit 1, Mecklenberg County, North Carolina

Date of application for amendment: August 27, 1999.

Brief description of amendment: The amendment approves a one-time extension of the surveillance frequency for Technical Specifications Surveillance Requirement (TSSR) 3.1.4.2 beyond the 25 percent extension allowed by TSSR 3.0.2 to the McGuire Nuclear Station, Unit 1. This license amendment is effective upon issuance and is to expire upon entering Mode 3 during Unit 1 startup following the Unit 1 End of Cycle 13 refueling outage.

Date of issuance: September 8, 1999.

Effective date: As of its date of issuance (September 8, 1999), and shall expire upon entering Mode 3 during startup, following the End of Cycle 13 refueling outage.

Amendment No.: Unit 1–186.

Facility Operating License No. NPF–9: Amendments revised the Technical Specifications.

Press release issued requesting comments as to proposed no significant hazards consideration: Yes, September 2, 1999, *Charlotte Observer.*

Comments received: No.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of North Carolina, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated September 8, 1999.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: September 13, 1999.

Brief description of amendments: The amendments revise the Technical Specifications TS 3.7.9, ''Control Room Area Ventilation System (CRAVS),'' to establish actions to be taken for an inoperable control room ventilation system due to a degraded control room pressure boundary. This revision approves changes that would allow up to 24 hours to restore the Control Room Pressure Boundary (CRPB) to operable status when two CRAVS trains are inoperable due to an inoperable CRPB in MODES 1, 2, 3, and 4. In addition, a Limiting Condition for Operation note would be added to allow the CRPB to be opened intermittently under administrative control without affecting CRAVS operability.

Date of issuance: September 22, 1999.

Effective date: As of the date of issuance and shall be implemented upon receipt.

Amendment Nos.: Unit 1—187; Unit 2—168.

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

Press release issued requesting comments as to proposed no significant hazards consideration: Yes, September 17, 1999, *Charlotte Observer.*

Comments received: No.

The Commission's related evaluation and the amendment, finding of emergency circumstances, consultation with the State of North Carolina, and final no significant hazards consideration determination are contained in a Safety Evaluation dated September 22, 1999.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland, this 29th day of September, 1999.

John A. Zwolinski,

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

[FR Doc. 99–25795 Filed 10–5–99; 8:45 am]

BILLING CODE 7590–01–P

SMALL BUSINESS ADMINISTRATION

Revocation of License of Small Business Investment Company

Pursuant to the authority granted to the United States Small Business Administration by the Final Order of the United States District Court for the District of Massachusetts, dated August 5, 1998, the United States Small Business Administration hereby revokes the license of Bever Capital Corporation, a Massachusetts corporation, to function as a small business investment company under the Small Business Investment Company License No. 01/01–0325 issued to Bever Capital Corporation on October 31, 1983 and said license is hereby declared null and void as of September 30, 1998.

Dated: September 30, 1999.

United States Small Business Administration.

Don A. Christensen,

Associate Administrator for Investment.

[FR Doc. 99–25981 Filed 10–5–99; 8:45 am]

BILLING CODE 8025–01–P

SMALL BUSINESS ADMINISTRATION

Revocation of License of Small Business Investment Company

Pursuant to the authority granted to the United States Small Business Administration by the Windup Order of the United States District Court for the Southern District of New York, dated June 4, 1999, the United States Small Business Administration hereby revokes the license of Diamond Capital Corporation, a New York corporation, to function as a small business investment company under the Small Business Investment Company License No. 02/02–0510 issued to Diamond Capital Corporation on January 21, 1988 and said license is hereby declared null and void as of September 30, 1999.

Dated: September 30, 1999.

United States Small Business Administration.

Don A. Christensen,

Associate Administrator for Investment.

[FR Doc. 99–25985 Filed 10–5–99; 8:45 am]

BILLING CODE 8025–01–P

SMALL BUSINESS ADMINISTRATION

Revocation of License of Small Business Investment Company

Pursuant to the authority granted to the United States Small Business Administration by the Final Order of the United States District Court for the Southern District of New York, dated August 24, 1998, the United States Small Business Administration hereby revokes the license of Everlast Capital Corporation a New York corporation, to function as a Small Business Investment Company under the Small Business Investment Company License No. 02/02–5468 issued to Everlast Capital Corporation on July 30, 1984 and said license is hereby declared null and void as of September 30, 1998.

Dated: September 30, 1999.