publication of a notice of proposed action and an opportunity for hearing or a notice of hearing is not warranted. Notice is hereby given of the right of interested persons to request a hearing on whether the action should be rescinded or modified. Also in connection with this action, the Commission performed an Environmental Assessment and determined that a Finding of No Significant Impact was appropriate for this action.

Further Information: The NRC has prepared a Safety Evaluation Report (SER) that documents the information that was reviewed and NRC's conclusion. In accordance with 10 CFR 2.390 of NRC's ''Rules of Practice,'' final NRC records and documents regarding this proposed action including the amendment request dated May 23, 2005, and the SER are publically available in the records component of NRC's Agencywide Documents Access and Management System (ADAMS). These documents may be inspected at NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* These documents may also be viewed electronically on the public computers, located at the NRC Public Document Room (PDR), O1F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852.

The PDR reproduction contractor will copy documents for a fee. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 17th day of March 2006.

For the Nuclear Regulatory Commission.

Jill S. Caverly,

Project Manager, Licensing Section, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E6–4445 Filed 3–27–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act; Notice of Meetings

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of March 27, April 3, 10, 17, 24, May 1, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of March 27, 2006

There are no meetings scheduled for the Week of March 27, 2006.

Week of April 3, 2006—Tentative

Monday, April 3, 2006

3:55 p.m.—Affirmation Session (Public Meeting) (Tentative).

a. USEC, Inc. (American Centrifuge Plant); Geoffrey Sea appeal of LBP–05–28 (Tentative).

b. USEC, Inc. (American Centrifuge Plan)—Appeal of LBP–05–28 by Portsmouth/Piketon Residents for Environmental Safety and Security (PRESS) (Tentative).

c. Hydro Resources, Inc.—Petition for Review of Partial Initial Decision on Phase II Cultural Resource Challengers (Tentative).

Week of April 10, 2006—Tentative

There are no meetings scheduled for the Week of April 10, 2006.

Week of April 17, 2006—Tentative

There are no meetings scheduled for the Week of April 17, 2006.

Week of April 24, 2006—Tentative

There are no meetings scheduled for the Week of April 24, 2006.

Monday, April 24, 2006

2 p.m.—Meeting with Federal Energy Regulatory Commission (FERC), FERC Headquarters, 888 First St., NE., Washington, DC 20426, Room 2C (Public Meeting).

Wednesday, April 26, 2006

1 p.m.—Discussion of Management Issues (closed—ex. 2).

Thursday, April 27, 2006

1:30 p.m.—Meeting with Department of Energy (DOE) on New Reactor Issues (Public Meeting).

This meeting will be webcast live at the Web address *http://www.nrc.gov.*

Week of May 1, 2006—Tentative

Tuesday, May 2, 2006

9:30 a.m.—Briefing on Status of Emergency Planning Activities—Morning Session (Public Meeting) (Contact: Eric Leeds, 301–415–2334).

1 p.m.—Briefing on Status of Emergency Planning Activities—Afternoon Session (Public Meeting).

This meeting will be webcast live at the Web address *http://www.nrc.gov.*

Wednesday, May 3, 2006

9 a.m.—Briefing on Status of Risk-Informed, Performance-Based Regulation (Public Meeting) (Contact: Eileen McKenna, 301–415–2189).

This meeting will be webcast live at the Web address *http://www.nrc.gov.*

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*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Michelle Schroll, (301) 415–1662.

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The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g., braille, large print), please notify the NRC's Disability Program Coordinator, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by e-mail at *DLC@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: March 23, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–3031 Filed 3–24–06; 1:15 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the

Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 3, 2006 to March 16, 2006. The last biweekly notice was published on March 14, 2006 (71 FR 13169).

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. Within 60 days after the date of publication of this notice, 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.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, 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 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, 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 general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include 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/requestor to relief. A petitioner/requestor who fails to satisfy 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.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, 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 by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov*; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: December 1, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.6.4.1, ''Secondary Containment.'' Specifically, the change would modify Surveillance Requirements (SRs) 3.6.4.1.4 and 3.6.4.1.5 to clarify their intent with respect to secondary containment boundary integrity. The change is submitted in accordance with the TS Task Force Traveler 322–A, Revision 2, ''Secondary Containment and Shield Building Boundary Integrity SRs.''

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

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

Response: No. This change involves an administrative clarification to reflect the original intent of the Technical Specifications. There is no impact on the availability or capability of the secondary containment or Standby Gas Treatment (SGT) system as a result of the proposed change. Both the secondary containment and SGT system are considered accident-mitigating equipment and are not initiators of any previously evaluated accidents. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated. Additionally, the proposed change does not alter the secondary containment or SGT systems' performance measures or their ability to perform their accident mitigation functions.

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

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

Response: No. The proposed changes to the wording of TS SRs 3.6.4.1.4 and 3.6.4.1.5 clarify that only one SGT subsystem is required to ensure the requirements of TS 3.6.4.1 are met. The proposed change does not alter the parameters within which the plant is operated. There are no new system operating conditions or performance measures introduced by this proposed change that will affect the secondary containment and SGT systems' protective or mitigative functions. The proposed changes will not alter the methods in which equipment is operated or tested. No new accident scenarios or assumptions, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

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

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

Response: No. Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms or actions. The proposed change does not impact any of these margins of safety parameters. This change involves an administrative clarification to reflect the original intent of the TS. There is no adverse effect on the operability or design requirements of the secondary containment or SGT system. The equipment will continue to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its intended safety function. There is no impact on the plant safety analyses.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendments request: February 27, 2006.

Description of amendments request: The amendment would revise Technical Specification 4.2.1, ''Fuel Assemblies,'' to allow fuel with advanced cladding material to be installed in the core for Cycle 19 only at Unit No. 1 or Cycle 17 only at Unit No. 2. Advanced cladding material from Framatome-ANP may be used in up to 2 lead test assemblies, and advanced cladding material from Westinghouse may be used in up to 2 lead test assemblies.

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.

Calvert Cliffs Technical Specification 4.2.1, Fuel Assemblies, states that fuel rods are clad with either Zircaloy or ZIRLO™. Calvert Cliffs Nuclear Power Plant, Inc. proposes to re-insert up to four fuel assemblies into Calvert Cliffs Unit 1 or Unit 2 that have some fuel rods clad in zirconium alloys that do not meet the definition of Zircaloy or ZIRLO™. A temporary exemption to the regulations has also been requested to allow these fuel assemblies to be re-inserted into Unit 1 or Unit 2. The proposed change to the Calvert Cliffs Technical Specifications will allow the use of cladding materials that are not Zircaloy or ZIRLO™ for one fuel cycle once the temporary exemption is approved. The proposed change to the Technical Specification is effective only as long as the temporary exemption is effective. The addition of what will be an approved temporary exemption for Unit 1 or Unit 2 to Technical Specification 4.2.1 does not change the probability or consequences of an accident previously evaluated.

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

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

The proposed change does not add any new equipment, modify any interfaces with existing equipment, change the equipment's function, or change the method of operating the equipment. The proposed change does not affect normal plant operations or configuration. Since the proposed change does not change the design, configuration, or operation, it could not become an accident initiator.

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

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

The proposed change will add an approved temporary exemption to the Calvert Cliffs Technical Specifications allowing the installation of up to four lead fuel assemblies. The assemblies use advanced cladding materials that are not specifically permitted by existing regulations or Calvert Cliffs' Technical Specifications. A temporary exemption to allow the installation of these assemblies has been requested. The addition of an approved temporary exemption to Technical Specification 4.2.1 is an administrative change to allow the installation of the lead fuel assemblies under the provisions of the temporary exemption. The license amendment is effective only as long as the exemption is effective. This amendment does not change the margin of safety since it only adds a reference to an approved, temporary exemption to the Technical Specifications.

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 amendments request involves no significant hazards consideration.

Attorney for licensee: Carey Fleming, Sr. Counsel—Nuclear Generation, Constellation Generation Group, LLC, 750 East Pratt Street, 17th floor, Baltimore, MD 21202.

NRC Branch Chief: Richard J. Laufer.

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 amendment request: January 5, 2005, supplemented November 21, 2005.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS) 5.5.19.b, TS 5.5.19.c, and TS Surveillance Requirement (SR) 3.8.1.9. TS 5.5.19.b currently requires verification that a Lee Combustion Turbine (LCT) can supply the equivalent of one Unit's maximum safeguard loads, plus two Units' Mode 3 loads, when connected to the system grid every 12 months. In the proposed amendments, this requirement would be more clearly specified as, ''Verify an LCT can supply equivalent of one Unit's Loss of Coolant Accident (LOCA) loads plus two Unit's Loss of Offsite Power (LOOP) loads when connected to system grid every 12 months.'' TS 5.5.19.b and SR 3.8.1.9 would be revised for consistency.

This notice supersedes the notice published in the **Federal Register** on February 15, 2005 (70 FR 7764).

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

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

Duke proposes to revise TS 5.5.19.b to clarify the Lee Combustion Turbine (LCT) testing requirements. Duke proposes to revise TS 5.5.19.c and TS 3.8.1 Surveillance Requirement (SR) 3.8.1.19 to be consistent with the proposed change to TS 5.5.19.b. The proposed change makes the wording of the test requirement consistent with the UFSAR [Updated Final Safety Analysis Report]. LCT testing has no impact on the probability of an accident analyzed in the UFSAR. The LCT can be credited to mitigate the consequences of an accident analyzed in the UFSAR. However, this clarification of LCT testing requirements has no impact on its ability to mitigate the consequences of an accident. As such, the proposed LAR [license amendment request] does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Duke proposes to revise TS 5.5.19.b to clarify the Lee Combustion Turbine (LCT) testing requirements. Duke proposes to revise TS 5.5.19.c and TS 3.8.1 SR 3.8.1.9 to be consistent with the proposed change to TS 5.5.19.b. The proposed change makes the wording of the test requirement consistent with the UFSAR. These changes do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment 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 TS change does not unfavorably affect any plant safety limits, set points, or design parameters. The changes also do not unfavorably affect the fuel, fuel cladding, RCS [reactor coolant system], or containment integrity. Therefore, the proposed TS change, which clarifies TS requirements associated with the LCT testing program, does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: March 13, 2006.

Description of amendment request: The proposed amendments would make changes to the technical specifications (TS) for LaSalle County Station (LSCS), Units 1 and 2. Surveillance Requirement (SR) 3.7.3.1 verifies the cooling water temperature supplied to the plant from the core standby cooling system (CSCS) pond (i.e., the ultimate heat sink (UHS)) is ≤ 100 °F. Currently, if the temperature of the cooling water supplied to the plant from the CSCS pond is > 100 °F, the UHS must be declared inoperable in accordance with TS 3.7.3. TS 3.7.3, Required Action B.1, requires that both units be placed in Mode 3 within 12 hours and Required Action B.2 requires that both units be placed in Mode 4 within 36 hours.

Prolonged hot weather in the area during the summer months, in conjunction with high humidity during the daytime, minimal cooling at night and little precipitation, has resulted in sustained elevated cooling water temperature supplied to the plant from the CSCS pond. This license amendment is being requested to increase the temperature limit of the cooling water supplied to the plant from the CSCS pond to ≤ 101.5 °F by reducing the temperature measurement uncertainty by replacing the existing thermocouples with higher precision temperature measuring equipment. Should the UHS indicated temperature exceed 101.5 °F, Required Action B.1 would be entered and both units would be placed in Mode 3 within 12 hours and Mode 4 within 36 hours.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change will allow the indicated temperature of the cooling water supplied to the plant from the CSCS pond to be increased to ≤ 101.5 °F based on reducing the temperature measurement uncertainty by replacing the existing thermocouples with higher precision temperature measuring equipment.

Analyzed accidents are assumed to be initiated by the failure of plant structures, systems, or components. An inoperable UHS is not considered as an initiator of any analyzed events. As such, there is not a significant increase in the probability of a previously evaluated accident. Allowing the UHS to operate at a higher allowable indicated temperature, but still within the design limits of the equipment it supplies, will not affect the failure probability of that equipment. The current heat analyses calculations of record for LSCS, Units 1 and 2, assume a UHS temperature of 100 °F and post-accident peak inlet temperature of 104 °F. The proposed temperature increase is based solely on a reduction of the existing instrument loop uncertainty value. The current analysis bounds the proposed change. This higher allowable indicated temperature does not impact the LOCA [loss-of-coolant accident] Peak Clad Temperature Analysis, LOCA Containment Analysis or the non-LOCA analyses; therefore, continued operation with a UHS temperature > 100 °F but ≤ 101.5 °F will not increase the consequences of an accident previously evaluated in the UFSAR.

Based on the above information, the increase in the allowable indicated temperature of the cooling water supplied to the plant from the UHS to ≤ 101.5 °F by reducing the existing instrument loop uncertainty value has no effect on the result of the design basis event and will continue to allow each required heat exchanger to perform its safety function. The heat exchangers will continue to provide sufficient cooling for the heat loads during the most severe 30-day period.

Based on the above information, increasing the allowable indicated temperature of the cooling water supplied to the plant from the CSCS pond from ≤ 100 °F to ≤ 101.5 °F by reducing the instrument uncertainty value has no impact on any analyzed accident; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves replacing the presently installed thermocouples with higher accuracy temperature measurement equipment. This proposed action will not alter the manner in which equipment is operated, nor will the functional demands on credited equipment be changed. No alteration in the procedures that ensure the units remain within analyzed limits is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. Raising the UHS temperature limit does not introduce any new or different modes of plant operation, nor does it affect the operational characteristics of any safety-related equipment or systems; as such, no new failure modes are being introduced. The proposed action reduces the instrument uncertainty value but does not alter assumptions made in the safety analysis.

Increasing the allowable indicated temperature of the cooling water supplied to the plant from the CSCS pond from ≤ 100 °F to ≤ 101.5 °F has no impact on safety related systems. The plant is designed such that the RHR [residual heat removal] pumps on the unit undergoing the LOCA/LOOP [loss of offsite power] conditions would start upon the receipt of a signal, and would load onto their respective Emergency Diesel Generators emergency bus during the LOOP event. The increase in the allowable indicated temperature of the cooling water supplied to the plant from the CSCS pond will not require operation of additional RHR pumps; therefore, system operation is unaffected by the proposed change in the UHS temperature limit.

Based on the above information, 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 proposed change allows an increase in the allowable indicated temperature of the cooling water supplied to the plant from the CSCS pond to ≤ 101.5 °F. The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed action does not impact these factors as the analyzed peak inlet temperature of the UHS is unaffected based on the improved instrument uncertainty of the new high precision temperature measurement instrumentation. No setpoints are affected, and no other change is being proposed in the plant operational limits as a result of this change. All accident analysis assumptions and conditions will continue to be met. Adequate design margin is available to ensure that the required margin of safety is not significantly reduced.

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: February 14, 2006.

Description of amendment request: The proposed amendment would revise the frequency of the Mode 5 Intermediate Range Monitoring (IRM) Instrumentation CHANNEL FUNCTIONAL TEST contained in Technical Specification (TS) 3.3.1.1 from 7 days to 31 days. The methodology used for the IRM drift analysis is based upon guidance contained in Generic Letter 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle,'' and Electric Power Institute Report TI–103335, ''Guidance for Instrument Calibration Extension/Reduction Programs.''

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 Technical Specifications (TS) change involves an increase in the Mode 5 CHANNEL FUNCTIONAL TEST interval for RPS [Reactor Protection System] IRM channels from 7 days to 31 days. The IRM system is used for event mitigation. The failure of an IRM does not initiate an accident or transient event. The proposed TS change does not alter the design or function of the IRM system for no physical changes are being made to the plant. Evaluation of the proposed testing interval change demonstrated that the availability of IRMs to mitigate the consequences of a control rod withdrawal event at low power levels are not significantly affected based on the effectiveness of other, required TS surveillance testing that is performed, the availability of redundant systems and equipment, and the high reliability of the IRM equipment.

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.

The proposed TS change involves an increase in the Mode 5 IRM CHANNEL FUNCTIONAL TEST interval from 7 clays [days] to 31 days. Existing TS testing requirements ensure the operability of the IRMs. The proposed TS change does not introduce any failure mechanisms of a different type than those previously evaluated, since no physical changes to the plant are being made. No new or different equipment is being installed, and no installed equipment is being operated in a different manner. As a result, no new failure modes are introduced. In addition, the manner in which surveillance tests are performed remains unchanged.

Therefore, the proposed TS 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 proposed TS change involves an increase in the Mode 5 CHANNEL FUNCTIONAL TEST interval for RPS IRM channels from 7 days to 31 days. There is expected to be no impact on system operability, based upon the performance of the more frequent Channel Checks, Control Room monitoring when the IRMs are in use, and the overall IRM reliability.

Furthermore, a historical review of surveillance test results and associated maintenance records did not indicate evidence of any failure that would invalidate the above conclusions.

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

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

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Mindy S. Landau, Acting.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: December 16, 2005.

Description of amendment request: The proposed change to Technical Specification (TS) Surveillance Requirement (SR) 4.1.4d relocates the SR for testing the core spray header differential pressure (ΔP) instrumentation to licensee-controlled documents. TS SR 4.1.4d currently requires that the core spray header ΔP instrumentation be periodically tested such that a check of each sensor is performed at least once each day and each channel is calibrated and tested at least once every 3 months. The proposed change will allow these SRs to be placed in licensee-controlled documents where future changes will be made pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59. The functional description of the core spray header ΔP instrumentation will also be relocated from the TS Bases to licensee-controlled documents consistent with the proposed TS 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, which is presented below:

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

Response: No.

The proposed changes are limited to the relocation of selected instrumentation requirements. The proposed relocated requirements were determined not to meet the 10 CFR 50.36 screening criteria for retention in the TSs and will be maintained in licensee-controlled documents in accordance with the provisions of 10 CFR 50.59. The proposed changes do not introduce any new modes of plant operation, make any physical changes to the plant, or alter any operational setpoints which could degrade the performance of any safety system assumed to function in the accident analysis. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No. The proposed changes do not introduce any new modes of plant operation, make any physical changes to the plant, or alter any operational setpoints which could create new accident initiators or failure mechanisms. The proposed changes are limited to the relocation of selected instrumentation requirements, and will have no impact on the accident assumptions and initial conditions as previously analyzed in the UFSAR [Updated Final Safety Analysis Report]. 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 proposed change involve a significant reduction in a margin of safety?

Response: No. The proposed changes are consistent with the Improved Standard TSs (NUREG–1433, Rev. 3) and will have no impact on the instrumentation setpoints, logic, or functional requirements as described in the TSs, TS Bases, and UFSAR. The proposed relocated requirements were determined to not meet the 10 CFR 50.36 screening criteria for retention in the TSs. Thus, the relocated requirements will be maintained in accordance with 10 CFR 50.59 as required. Accordingly, the proposed relocated requirements will not degrade the quality or performance of any safety system assumed to mitigate an accident or assure operation within the safety 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 request involves no significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Richard J. Laufer.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of amendment request: February 28, 2006.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specification (TS) Surveillance Requirements (SRs) 3.8.4.7 and 3.8.4.8 to clarify that diesel generator ''E'' (DG E) electrical power subsystem testing does not require a mode restriction when the DG E diesel is not required to be OPERABLE.

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

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

Response: No. Performance of TS required SRs are not initiators to any accident sequences analyzed in the Final Safety Analysis Report (FSAR). The changes do not involve any physical change to structures, systems, or components, (SSCs) and do not alter the method of operation or control of SSCs. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents are unaffected by these changes. No additional failure modes or mechanisms are being introduced and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed Technical Specification (TS) ensures that the DC [direct current] distribution system and supported equipment functions remain capable of performing the function as described in the FSAR. Therefore, the mitigative functions supported by the system will continue to provide the protection assumed by the analysis.

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

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

Response: No. The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints, at which protective or mitigative actions are initiated, affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the FSAR. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis.

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

Response: No. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change is acceptable because performance of SRs on equipment not require[d] to be OPERABLE and isolated from the OPERABLE plant equipment cannot affect any margin of safety. Therefore, the plant response to analyzed events will continue to provide the margin of safety assumed by the analysis.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer

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

Date of amendment requests: March 10, 2006.

Description of amendment requests: The licensee requests the Nuclear Regulatory Commission consent to the transfer of the City of Anaheim's 3.16 percent undivided ownership interest in SONGS 2 and 3 to Southern California Edison, excluding Anaheim's interest in its spent fuel and the SONGS 2 and 3 independent spent fuel storage installation.

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

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

Response: No. The amendments do not involve any change in the design, configuration, or operation of the nuclear plant. All Limiting Conditions for Operation, Limiting Safety System Settings, and Safety Limits specified in the Technical Specifications remain unchanged. SCE will continue to be the licensed operator of the units.

The technical qualifications of SCE to carry out its exclusive responsibilities under the operating licenses, as amended, will remain unchanged. Personnel engaged in operation, maintenance, engineering, assessment, training, and other related services are not changed. The SCE officers and executives currently responsible for the overall safe operation of the nuclear plants will continue in the same capacity.

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

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

Response: No. The amendments do not involve any change in the design, configuration, or operation of the nuclear plant. The current plant design and design bases will remain the same. The current plant safety analyses, therefore, remain complete and accurate in addressing the design basis events and in analyzing plant response and consequences.

The Limiting Conditions for Operation, Limiting Safety System Settings, and Safety Limits specified in the Technical Specifications are not affected by the change. As such, the plant conditions for which the design basis accident analyses were performed remain valid.

The amendments do not introduce a new mode of plant operation or new accident precursors, do not involve any physical alterations to plant configurations, or make changes to system set points that could initiate a new or different kind of accident.

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

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

Response: No. The amendments do not involve a change in the design, configuration, or operation of the nuclear plants. The change does not affect either the way in which the plant structures, systems, and components perform their safety function, or their design and licensing basis.

Plant safety margins are established through Limiting Conditions for Operation, Limiting Safety System Settings, and Safety Limits specified in the Technical Specifications. Because there is no change to the physical design of the plant, there is no change to any of these margins.

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 requests involve no significant hazards consideration.

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

NRC Branch Chief: David Terao.

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

Date of amendment request: September 19, 2005.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Limiting Conditions for Operation (LCO) 3.3.1, "Reactor Trip system (RTS) Instrumentation" and TS Surveillance Requirements (SR) 3.2.4.2, "Quadrant Power Tilt Ration (QPTR)" to avoid confusion as to when a flux map for QPTR is required.

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

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

No. The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be release offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated

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

No. The proposed changes do not result in a change in the manner in which the RTS and ESFAS provide plant protection. The RTS and ESFAS will continue to have the same set points after the proposed changes are implemented. There are no design changes associated with the license amendment.

The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

No. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis.

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

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: January 10, 2006 (TS–453).

Description of amendment request: The proposed amendment would specify the methodology used for determining, setting, and evaluating as-found setpoints for those drift susceptible instruments, which are either necessary to ensure compliance with a Safety Limit or critical in ensuring the fuel peak cladding temperature acceptance criteria of 10 CFR 50.46 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. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. Including references to TVA's methodology for determining, setting, and evaluating as-found instrument setpoints in the TS is an administrative change. There will be no change to the manner in which Safety Limits, Analytical Limits, or Allowable Values are determined. No changes are proposed in the manner in which the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS), Reactor Core Isolation Cooling (RCIC), or Primary Containment Isolation systems provide plant protection or which create new modes of plant operation.

The proposed request will not affect the probability of any event initiators. There will be no degradation in the performance of, or 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 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No. There are no hardware changes nor are there any changes in the method by which any plant system performs a safety function. This request does not affect the normal method of plant operation. The proposed amendment does not introduce new equipment, which could create a new or different kind of accident.

No new external threats, release pathways, or equipment failure modes are created. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this request. Therefore, the implementation of the proposed amendment will not create a possibility for an accident of a new or different type than those previously evaluated.

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

Response: No. Including references to TVA's methodology for determining, setting, and evaluating as-found instrument setpoints in the TS is an administrative change. No changes are proposed in the manner in which the RPS, ECCS, RCIC, or Primary Containment Isolation systems satisfy the Updated Final Safety Analysis Report requirements for accident mitigation or unit safe shutdown. There will be no change to Safety Limits, Analytical Limits, Allowable Values, or post-Loss Of Coolant Accident peak clad temperatures. For these reasons, 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 three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: February 6, 2006.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.7. The changes are consistent with Nuclear Regulatory Commission approved Industry/Technical Specification Task Force (TSTF) standard TS change TSTF–373, Revision 4. The availability of this TS improvement was published in the **Federal Register** on May 4, 2005 (70 FR 23252), as part of the consolidated line item improvement process.

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

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

Response: No. The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. Entrance into Actions or delaying entrance into Actions is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on the delay time allowed before declaring a TS supported system inoperable and taking its Conditions and Required Actions are no different than the consequences of an accident under the same plant conditions while relying on the existing TS supported system Conditions and Required Actions. Therefore, the consequences of an accident previously evaluated are not significantly increased by this change. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind

of accident from any accident previously evaluated?

Response: No. The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. The proposed change restores an allowance in the pre-ISTS conversion TS that was unintentionally eliminated by the conversion. The pre-ISTS TS were considered to provide an adequate margin of safety for plant operation, as does the post-ISTS conversion TS. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Michael L. Marshall, Jr.

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of amendment request: February 15, 2006.

Description of amendment request: The amendment would revise the Technical Specifications (TS) to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The proposed amendment includes changes to the TS definition of Leakage, TS 3.4.6.2, ''Reactor Coolant System, Operational Leakage,'' TS 3.4.5, ''Steam Generator (SG) Tube Integrity,'' and adds TS 6.8.4.k, ''Steam Generator (SG) Program,'' and TS 6.9.1.16, ''Steam Generator Tube Inspection Report.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97–06, ''Steam Generator Program Guidelines.''

The amendment would also delete License Condition 2.C.8 Item b. This License Condition references the licensee's letters from 1997 that contain commitments associated with NRC Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,'' and the application of voltage-based alternate repair criteria to the steam generators. The licensee has concluded that the provisions and requirements of the proposed TS changes bound the commitments identified in the existing License Condition.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated August 31, 2005.

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

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

The proposed change requires a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as a main steamline break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: December 15, 2005.

Description of amendment request: The amendment would revise the Technical Specifications (TS) to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The proposed amendment includes:

—Revised TS definition of Leakage,
—Revised TS 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,''
—Added new TS 3.4.17, ''Steam Generator Tube Integrity,''
—Revised TS 5.7.2.12, ''Steam Generator (SG) Tube Surveillance Program,'' and
—Revised TS 5.9.9, ''SG Tube Inspection Report.''

The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97–06, ''Steam Generator Program Guidelines.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated December 15, 2005.

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

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

The proposed change requires a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as a main steamline break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: December 12, 2005.

Brief description of amendments: The amendments requested would revise Technical Specification (TS) 3.3.1, ''RTS [Reactor Trip System] Instrumentation,'' Surveillance Requirements (SRs) 3.3.1.2 and SR 3.3.1.3.

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

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The Reactor Trip System (RTS) Instrumentation will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the Final Safety Analysis Report (FSAR) are not adversely affected because the change to the daily surveillance for the normalization of the Nuclear Instrumentation System (NIS) Power Range and Nitrogen-16 (N–16) Power Monitor indications assures the conservative response of the channel even at reduced power levels.

The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of, or 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.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

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

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No. There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating parameters. No performance requirements or response time limits will be affected.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No. The proposed changes require a revision to the criteria for implementation of NIS Power Range and N–16 Power Monitor indication adjustments based on secondary power calorimetric calculations; however, the changes do not eliminate any RTS surveillances or alter the frequency of surveillances required by the TS. The revision to the criteria for implementation of the daily surveillance will remove a requirement for normalization of the NIS Power Range and N–16 Power Monitor indications at reduced power conditions that could result in safety performance outside the bounds of the safety analyses. Therefore, the Nominal Trip Setpoints and Allowable Values for the Reactor Trip System functions, as specified in the TS and related Bases, as well as the safety analysis limits assumed in the transient and accident analyses, are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety systems 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, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequences are unaffected by this proposed change.

The imposition of appropriate surveillance testing requirements will not reduce any margin of safety since the changes will assure that safety analysis assumptions on equipment operability are verified on a periodic frequency.

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

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

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

NRC Branch Chief: David Terao.

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.

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

Date of amendment request: February 14, 2006.

Description of amendment request: Revise the Technical Specifications regarding the Containment Ventilation System to allow additional corrective actions for inoperable containment purge supply and exhaust valves.

Date of publication of individual notice in the ***Federal Register:*** March 1, 2006 (71 FR 10566).

Expiration date of individual notice: March 15, 2006.

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 (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

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 20, 2005, as supplemented by letter dated November 2, 2005.

Brief description of amendment: This amendment revises the footnotes in Tables 3.4–2 and 4.4–3 of Technical Specification (TS) 3/4.4.7 by increasing the temperature limit above which (1) reactor coolant sampling and analysis for dissolved oxygen is required, and (2) when limit for dissolved oxygen, specified in TS 4.4.7, applies. This temperature limit will be increased from 180 °F to 250 °F.

Date of issuance: March 8, 2006.

Effective date: March 8, 2006.

Amendment No. 120.

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

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59084). The supplemental letter provided clarifying information that was within the scope of the initial notice and 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 March 8, 2006.

No significant hazards consideration comments received: No.

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 1, 2005, as supplemented by letters dated December 22, 2005, and January 23, 2006.

Brief description of amendment: This amendment revises the technical specification (TS) requirements for pressurized-water reactor Boraflex fuel storage racks and adds TS requirements for fuel storage pool boron concentration. Specifically, the amendment (1) adds a new TS 3/4.7.14, ''Fuel Storage Pool Boron Concentration,'' with a Limiting Condition for Operation that requires a fuel pool boron concentration of at least 2000 ppm at all times, (2) revises and reformats TS 5.6.1 to specify the design features and fuel storage limitations in accordance with the categorization of spent fuel storage racks in various spent fuel pools, and (3) revises TS 5.3.1 to remove the cross-reference to TS 5.6.1.b.

Date of issuance: March 10, 2006.

Effective date: March 10, 2006.

Amendment No. 121.

Facility Operating License No. NPF–63: Amendment revises the TS.

Date of initial notice in ***Federal Register:*** November 8, 2005 (70 FR 67745). The supplemental letters provided additional information that was within the scope of the initial notice and 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 March 10, 2006.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket Nos. 50–336 and 50–423, Millstone Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of application for amendments: February 25, 2005.

Brief description of amendments: The amendments made various administrative changes to the Technical Specifications (TSs).

Date of issuance: March 16, 2006.

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

Amendment Nos.: 291 and 229.

Facility Operating License Nos. DPR–65 and NPF–49: The amendments revised the TSs.

Date of initial notice in ***Federal Register:*** March 29, 2005 (70 FR 15942).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 16, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: December 17, 2004.

Brief description of amendments: The amendments revised Appendix B, Environmental Protection Plan (non-radiological), of the LaSalle County Facility Operating Licenses.

Date of issuance: March 8, 2006.

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

Amendment Nos.: 176/162.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Environmental Protection Plan.

Date of initial notice in ***Federal Register:*** April 12, 2005 (70 FR 19115).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 15, 2005.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.3.2.2, ''Feedwater System and Main Turbine High Water Level Trip Instrumentation,'' to reflect a design change in the instrumentation logic that trips the three feedwater pumps and main turbine.

Date of issuance: March 9, 2006.

Effective date: As of the date of issuance and shall be implemented prior to start-up from the spring 2006 refueling outage for Unit 2 and prior to start-up from the spring 2007 refueling outage for Unit 1.

Amendment Nos.: 330/225.

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

Date of initial notice in ***Federal Register:*** August 30, 2005 (70 FR 51381).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August, 20, 2004, as supplemented by letters dated June 16 and December 6, 2005.

Brief description of amendment: The amendment revised TS 3/4.8.1.1,''A.C. Sources—Operating,'' by deleting Surveillance Requirement (SR) 4.8.1.1.2.d.4, which requires verification that the emergency diesel generator auto-connected loads do not exceed the 2000-hour load limit. In addition, the amendment revised TS 4/3.8.1.2, ''A.C. Sources—Shutdown,'' to add exceptions to SR 4.8.1.2 when performed in Modes 5 and 6. As a result of discussions held on October 20, 2005, FENOC decided to withdraw the portion of the amendment request (LAR 01–0009) that requested clarification of SR 4.8.1.1.b.

Date of issuance: March 2, 2006

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

Amendment No.: 273.

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

Date of initial notice in ***Federal Register:*** September 28, 2004 (69 FR 57989).

The June 16 and December 6, 2005, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration 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 March 2, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 2, 2005, as supplemented by letters dated August 28, September 15, 2005, and January 12, 2006, and January 13, February 9, and February 28, 2006.

Brief description of amendment: This amendment revised the Technical Specifications (TSs) Section 2.1.1, ''Safety Limits—Reactor Core,'' and TS Section 2.2.1, ''Limiting Safety Settings—Reactor Protection System Setpoints.'' The amendment supports the use of the Framatome Mark B–HTP fuel design for Cycle 15, which is scheduled to begin following the refueling outage in March 2006.

Date of issuance: March 2, 2006

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

Amendment No.: 274.

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

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29796).

The August 28, September 15, 2005, and January 12, January 13, February 9, and February 28, 2006, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 25, 2005.

Brief description of amendment: The amendment revised the definitions of Channel Calibration, Channel Function Test, and Logic System Functional Test in accordance with the Technical Specification Task Force Traveler (TSTF)–205–A.

Date of issuance: March 10, 2006

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

Amendment No.: 217.

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

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59086).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 10, 2006

No significant hazards consideration comments received: No.

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

Date of amendment request: August 11, 2005.

Brief description of amendment: The amendment deleted the surveillance requirement (SR) of TS 2.10.2(9)b(iii) to verify shutdown margin every 8-hour shift during low power physics testing. This change made TS 2.10.2(9)b more consistent with SR 3.1.7 of NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants, Revision 3.'' In addition, the Containment Structural Tests Report has been deleted from TS 5.9.3c and several administrative and editorial changes were made.

Date of issuance: February 1, 2006.

Effective date: February 1, 2006 and shall be implemented within 60 days from the date of issuance.

Amendment No.: 237.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 27, 2005 (70 FR 56503)

The Commission's related evaluation of the amendment is contained in a safety evaluation dated February 1, 2006.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of application for amendments: June 27, 2005, as supplemented on December 1, 2005, and February 28, 2006.

Brief description of amendments: These amendments change the SSES 1 and 2 technical specifications for reactor protection system and control rod block instrumentation, oscillation power range monitor instrumentation, recirculation loops operating, shutdown margin test—refueling, and the core operating limits report. The amendments modify the power range neutron monitor system (PRNMS) by installation of the General Electric Nuclear Measurement Analysis and Control PRNMS. The modification of the PRNMS replaces analog technology with a digital upgrade.

Date of issuance: March 3, 2006

Effective date: As of the date of issuance and to be implemented prior to startup following the Cycle 14 refueling outage for Unit 1 and the Cycle 13 refueling outage for Unit 2.

Amendment Nos.: 230 and 207.

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

Date of initial notice in ***Federal Register:*** September 13, 2005 (70 FR 54088).

The supplements dated December 1, 2005, and February 28, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 3, 2006.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2 (SSES 2), Luzerne County, Pennsylvania

Date of application for amendment: March 18, 2005, as supplemented on February 28, 2006.

Brief description of amendment: The amendment revises the SSES 2 Technical Specification 3.3.8.1, ''Loss of Power (LOP) Instrumentation,'' to (1) clarify that Condition A applies to the LOP instrumentation associated with both the Unit 1 and Unit 2 4.16 Kilovolt (kV) Engineered Safeguards System (ESS) buses since both the Unit 1 and Unit 2 buses are required to support Unit 2 operation, (2) add a new Condition B to allow the LOP instrumentation for two Unit 1 4.16kV ESS buses in the same division to be inoperable for up to 8 hours for the performance of Surveillance Requirement 3.8.1.19 on Unit 1. In addition, the amendment revises the SSES 2 TS 3.8.7, ''Distribution Systems—Operating,'' to (1) eliminate ''or more'' and the plural to ''subsystems'' such that the condition will read ''one Unit 1 AC [alternating current] electrical power distribution subsystem inoperable,'' and (2) add a new Condition D for two Unit 1 AC electrical power distribution subsystems inoperable.

Date of issuance: March 16, 2006.

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

Amendment No.: 208.

Facility Operating License No. NPF–22: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29800).

The supplement dated February 28, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 16, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 4, 2005.

Brief description of amendment: The amendment relocated the Transversing In-Core Probe (TIP) system Technical Specification (TS) to the Hope Creek Generating Station Updated Final Safety Analysis Report, as well as removed the note on the TIP system from the Reactor Protection System Instrumentation Surveillance Requirements table.

Date of issuance: March 8, 2006.

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

Amendment No.: 164.

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

*Date of initial notice in **Federal Register:*** March 15, 2005 (70 FR 12750).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2006.

No significant hazards consideration comments received: No.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: April 29, 2005, as supplemented on July 1 and November 21, 2005.

Brief description of amendment: The amendment revised Technical Specification 3.7.3, ''Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs),'' to allow the use of the main feedwater isolation valves in lieu of the MFPDVs to provide isolation capability to the steam generators in the event of a steam line break.

Date of issuance: March 16, 2006

Effective date: As of the date of issuance to be implemented prior to startup from the fall 2006 refueling outage.

Amendment No.: 95.

Renewed Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 7, 2005 (70 FR 33218).

The July 1 and November 21, 2005, letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the ***Federal Register***.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 16, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 31, 2005.

Brief description of amendment: The amendment revises the Technical Specifications associated with steam generator tube integrity consistent with Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

Date of issuance: February 23, 2006.

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

Amendment No.: 306.

Facility Operating License No. DPR–77: Amendment revises the technical specifications.

*Date of initial notice in **Federal Register:*** November 22, 2005 (70 FR 70643).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 2006.

No significant hazards consideration comments received: No.

TXU Generation Company LP, Docket No. 50–446, Comanche Peak Steam Electric Station, Unit No. 2, Somervell County, Texas

Date of amendment request: April 27, 2005, as supplemented by letter dated July 20, 2005.

Description of amendment: The amendment revises the Technical Specifications to add Topical Report WCAP–13060–P–A to the list of NRC approved methodologies to be used at Comanche Peak Steam Electric Station, Unit 2.

Date of issuance: March 15, 2006.

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

Amendment No.: 123.

Facility Operating License No. NPF–89: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 8, 2005 (70 FR 67753).

No significant hazards consideration comments received: No.

TXU Generation Company LP, 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 6, 2004, as supplemented by letters dated September 16 and November 22, 2005.

Brief description of amendments: The amendments revised the Technical Specification 3.8.1, ''AC Sources—Operating,'' to remove mode restrictions on surveillance requirements.

Date of issuance: March 15, 2006.

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

Amendment Nos.: 124.

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

*Date of initial notice in **Federal Register:*** March 15, 2005 (70 FR 12751).

The supplements dated September 16 and November 22, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 20th day of March 2006.

For the Nuclear Regulatory Commission.

Edwin M. Hackett,

Deputy Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. 06–2908 Filed 3–27–06; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Excepted Service

AGENCY: Office of Personnel Management (OPM).

ACTION: Notice.

SUMMARY: This gives notice of OPM decisions granting authority to make appointments under Schedules A, B, and C in the excepted service as required by 5 CFR 6.6 and 213.103.

FOR FURTHER INFORMATION CONTACT: David Guilford, Center for Leadership