For More Information Contact: Vicky D'Onofrio, (202) 314–6410.

Vicky D'Onofrio,

Federal Register Liaison Office.

[FR Doc. 07–2050 Filed 4–20–07; 2:13 pm]

BILLING CODE 7533–01–M

NUCLEAR REGULATORY COMMISSION

Notice of Sunshine Act Meetings

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of April 23, 30, May 7, 14, 21, 28, 2007.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of April 23, 2007

Monday, April 23, 2007

2:30 p.m.
Discussion of Security Issues (Closed-Ex. 1).

Week of April 30, 2007—Tentative

There are no meetings scheduled for the Week of April 30, 2007.

Week of May 7, 2007—Tentative

Monday, May 7, 2007

1:25 p.m.
Affirmation Session (Public Meeting) (Tentative).
a. Consumers Energy Co. (Big Rock Point ISFSI); License Transfer Application (Tentative).

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

1:30 p.m.
Briefing on Office of Federal and State Materials and Environmental Management Programs (FSME) Programs, Performance, and Plans (Public Meeting) (Contact: George Deegan, 301–415–7834).

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

Week of May 14, 2007—Tentative

There are no meetings scheduled for the Week of May 14, 2007.

Week of May 21, 2007—Tentative

There are no meetings scheduled for the Week of May 21, 2007.

Week of May 28, 2007—Tentative

Tuesday, May 29, 2007

1:30 p.m.
NRC All Hands Meeting (Public Meeting) (Contact: Rickie Seltzer, 301–415–1728). Marriott Bethesda North Hotel, 5701 Marinelli Road, Rockville, MD 20852.

Wednesday, May 30, 2007

9:30 a.m.
Briefing on Results of the Agency Action Review Meeting (AARM)—Materials (Public Meeting).

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

10:15 a.m.
Discussion of Security Issues (Closed—Ex.1).

Thursday, May 31, 2007

9 a.m.
Briefing on Results of the Agency Action Review Meeting (AARM)—Reactors (Public Meeting).

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

* 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.

Additional Information

By a vote of 5–0 on April 19, 2007, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Security Issues (Closed-Ex. 1)'' be held April 23, 2007, and on less than one week's notice to the public.

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

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: April 19, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–2046 Filed 4–20–07; 11:09 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to 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 30, 2007 to April 12, 2007. The last biweekly notice was published on April 10, 2007 (72 FR 17944).

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, Rulemaking, Directives and Editing 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. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: March 22, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specifications to incorporate a revised limit for the variable low reactor coolant system pressure-temperature core protection safety limit. The revised limit is associated with the introduction of AREVA NP's Mark-B-HTP fuel design, which will require more restrictive Safety Limits and more restrictive Limiting Safety System Settings for the Reactor Protection System. The proposed limits are developed in accordance with the method described in the Nuclear Regulatory Commission (NRC)-approved Topical Report BAW–10179P–A, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.'' The revised limits will maintain the same magnitude of departure from nucleate boiling (DNB) protection.

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.

The proposed Technical Specification (TS) limits and reactor protection system (RPS) trip setpoints are developed in accordance with the methods and assumptions described in NRC-approved AREVA NP Topical Reports BAW–10179 P–A, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses'' and BAW–10187 P–A, ''Statistical Core Design for B&W-Designed 177 FA Plants.'' The core thermal-hydraulic code (LYNXT) and CHF correlation (BHTP) have been approved for use with these methods and the Mark-B-HTP fuel type. The proposed change preserves the design DNB Ratio safety criterion that there shall be at least a 95% [percent] probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling during normal operation or events of moderate frequency. The corresponding core-wide protection on a pin-by-pin basis is greater than 99.9%. The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the mixed core conditions at TMI Unit 1, where the Mark-B-HTP fuel design will be co-resident with earlier Mark-B fuel designs. The setpoint calculation methodology utilized, and the surveillance requirements established, are in accordance with approved industry standards and NRC criteria.

The proposed setpoint change does not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not alter any assumptions previously made in the radiological consequence evaluations, or affect mitigation of the radiological consequences of an accident previously evaluated.

Therefore, the proposed changes do 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.

The proposed TS limit and reactor protection system (RPS) trip setpoint provide a core protection safety limit and variable low pressure trip setpoint developed in accordance with NRC-approved methods and assumptions. No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function.

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 amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed RPS trip setpoint ensures core protection safety limits will be preserved during power operation. The proposed safety limit and setpoint are developed in accordance with NRC-approved methods and assumptions. The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the mixed core conditions at TMI Unit 1. The setpoint calculation methodology utilized, and the surveillance requirements established, are in accordance with approved industry standards and NRC criteria.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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. Brad Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Harold K. Chernoff.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: February 27, 2007.

Description of amendment request: The proposed license amendment would revise Technical Specification 4.2.1, Fuel Assemblies, to add a temporary exemption to allow the insertion of up to four lead fuel assemblies, which contain non-Zircaloy based cladding, into the Unit 1 core for one cycle of operation. These lead fuel assemblies are currently installed in the Unit 2 core under a previous exemption and are scheduled to be discharged during the 2007 refueling outage.

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

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 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 been requested to allow these fuel assemblies to be re-inserted into Unit 1. 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 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 accident 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 Nuclear Regulatory Commission (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: Carey Fleming, Esquire, Senior Counsel—Nuclear Generation, Constellation Generation Group, LLC, 750 East Pratt Street, 17th floor, Baltimore, MD 21202.

NRC Acting Branch Chief: John P. Boska.

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 amendment request: February 27, 2007.

Description of amendment request: The proposed license amendment would revise Technical Specification 5.6.5, Core Operating Limits Report (COLR), to add the supporting topical report (WCAP–15604–NP, Revision 2–A, ''Limited Scope High Burnup Lead Test Assemblies,'' September 2003) to the list of references. The topical report provides guidance for operation with a limited number of lead fuel assemblies to be irradiated to a higher burnup limit than currently allowed for Calvert Cliffs fuel 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. The licensee has determined that the proposed change:

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

The proposed change would modify the Calvert Cliffs Units 1 and 2 Technical Specification 5.6.5.b, Core Operating Limits Report by adding an approved topical report to the existing list of topical reports. The topical report provides the technical basis that supports irradiating a limited number of lead fuel assemblies to a higher burnup limit than currently approved for Calvert Cliffs. The proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident.

In the safety evaluation report approving the requested topical report (WCAP–15604–NP, Revision 2–A), the Nuclear Regulatory Commission concluded that it is acceptable for an individual power licensee to irradiate a limited number of lead fuel assemblies to a maximum burnup to 75 GWD/MTU [gigawatt days per metric ton of uranium] provided that certain conditions are met. Calvert Cliffs meets those required conditions. Because those required conditions are met and only a limited number of fuel assemblies are included in this change, the probability or consequences of an accident previously evaluated are not significantly increased.

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

2. 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 plant design, operation, or configuration, 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 accident previously evaluated.

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

The proposed change will add a reference to an approved topical report to allow a limited number of lead fuel assemblies to be irradiated to a higher burnup level than is currently allowed at Calvert Cliffs. The higher burnup limit has been evaluated and approved in the topical report being referenced. Calvert Cliffs conforms to the requirements of the topical report. The addition of an approved reference to the Technical Specifications is administrative in nature and has no impact on the margin of safety for any plant configuration or on system performance that is relied upon to mitigate the consequences on an accident.

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: Carey Fleming, Esquire, Senior Counsel—Nuclear Generation, Constellation Generation Group, LLC, 750 East Pratt Street, 17th floor, Baltimore, MD 21202.

NRC Acting Branch Chief: John P. Boska.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: February 2, 2007.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TS) to maintain hydrogen recombiners and hydrogen monitors. Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of Three Mile Island

(TMI) Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. The revised Title 10 of the Code of Federal Regulations (10 CFR) 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff published a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50374), on possible amendments to eliminate requirements regarding containment hydrogen recombiners and the removal of requirements from TS for containment hydrogen and oxygen monitors, including a model safety evaluation and model No Significant Hazards Consideration (NSHC) Determination, in accordance with 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 September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated February 2, 2007.

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 revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents.

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3 and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the severe accident management guidelines, the emergency plan, the emergency operating procedures, and site survey monitoring that support modification of emergency plan protective action recommendations.

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

Attorney for licensee: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Thomas H. Boyce.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: March 19, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) 3.8.1 entitled ''AC Sources-Operating'' to change the minimum Emergency Diesel Generator (EDG) output voltage acceptance criterion from 3740 to 3873 volts. Specifically, the proposed change would revise the Surveillance Requirements (SRs) 3.8.1.2, 3.8.1.7, 3.8.1.10, 3.8.1.11, 3.8.1.14, and 3.8.1.17.

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

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

The increase in the minimum EDG output voltage acceptance criterion value in TS 3.8.1 surveillance requirements does not adversely affect any of the parameters in the accident analyses. The change increases the minimum allowed EDG output voltage acceptance criterion to ensure that sufficient voltage is available to operate the required Emergency Safety Feature (ESF) equipment under accident conditions. The increase in the minimum allowed EDG output voltage in the TS surveillance requirements ensures that adequate voltage is available to support the assumptions made in the Design Bases Accident (DBA) analyses. DBA analyses assume that onsite standby emergency power will provide an adequate power source to operate safe shutdown equipment and to mitigate consequences of design bases accidents. This conservative change of the acceptance criterion enhances the testing requirements of the onsite emergency diesel generators and ensures the reliability of this power source. Changing the acceptance criterion does not affect the probability of evaluated accidents and it provides better assurance of EDG reliability in mitigating consequences of accidents. 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 an accident previously evaluated.

The change in the value of the minimum EDG output voltage acceptance criterion supports the assumptions in the accident analyses that sufficient voltage will be available to operate ESF equipment on the Class 1E buses when these buses are powered from the onsite emergency diesel generators. The maximum EDG output voltage of 4580 volts is not affected by this change. The change in the minimum EDG output voltage from 3740 to 3873 volts ensures the reliability of the onsite emergency power source. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed license amendment involves a change in the minimum EDG output voltage acceptance criterion in TS 3.8.1 surveillance requirements. The surveillance frequency and the different test requirements are unchanged. The change provides a better assurance that the onsite power source is able to satisfy the design requirements assumed in the accident analyses to safely shutdown the reactor and mitigate the consequences of design bases accidents. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: David G. Pettinari, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: November 8, 2006.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) Action and Surveillance Requirements (SRs) for instrumentation identified in TSs 3.3.1 and 3.3.2. In particular, the proposed amendment adds actions to address the inoperability of one or more automatic bypass removal channels; revises the terminology used in the notation of TS Tables 2.2–1 and 3.3–1 relative to the implementation and automatic removal of certain Reactor Protection System (RPS) trip bypasses; revises the frequency for performing surveillance of the automatic bypass removal function logic; and incorporates two administrative changes.

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

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

Response: No.

The proposed changes to Technical Specifications 2.2.1, 3.3.1 and 3.3.2 do not adversely impact structure, system, or component design or operation in a manner that would result in a change in the frequency of occurrence of accident initiation. The proposed technical specification changes do not involve accident initiators, do not change the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, and do not alter any conditions assumed in the plant accident analyses. The proposed amendment does not change the function or the manner of operation of the RPS or ESFAS [engineered safety features actuation system] trip bypass features. Adding actions to be taken for an inoperable automatic bypass removal function places additional restriction on plant operation in this condition and does not alter the setpoint or the logic of the operating bypasses and automatic bypass removals. Clarifying the frequency of the SR associated with testing the automatic bypass removal function does not alter the setpoint or the manner of operation of the operating bypasses and automatic bypass removals. More accurately reflecting the input process variable of the operating bypasses and automatic bypass removals of the affected reactor trips does not alter the setpoint nor the manner of operation of the operating bypasses and automatic bypass removals. With respect to the incorporation of the administrative changes, the proposed changes are spelling corrections and do not alter any of the requirements of the affected TS. Therefore, this change does not impact the consequences of any accident. Based on this discussion, the proposed amendment does not increase the probability or consequence of an accident previously evaluated.

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

Response: No.

No new or different accidents result from clarifying actions for an inoperable automatic bypass removal function, clarifying surveillance requirements for the automatic bypass removal function, and more accurately reflecting the parameter being measured for automatic bypass removal by referring to logarithmic power, the input process variable. The results of previously performed accident analyses remain valid. The proposed amendment does not introduce accident initiators or malfunctions that would cause a new or different kind of accident. The proposed amendments are administrative in nature and will not change the physical plant or the modes of plant operation defined in the facility operating license. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change does not alter the function or manner of operation of the operating bypasses and automatic bypass removals of the affected reactor trips. The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Branch Chief: Harold K. Chernoff.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: January 4, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specification for Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) for control rod operability, scram insertion times, and control rod accumulators.

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 extend the frequency and revise the methodology for testing control rod scram times, and identify a new category of ''slow'' control rods for assessing control rod operability. The frequency of control rod scram testing is not an initiator of any accident previously evaluated. The frequency of surveillance testing does not affect the ability to mitigate any accident previously evaluated, because the tested component is still required to be operable. The proposed test methodology is consistent with industry approved methods and ensures control rod operability requirements for the number and distribution of operable, slow, and stuck control rods continue to satisfy scram reactivity rate assumptions used in plant safety analysis.

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 proposed changes do not involve any physical alteration of the plant (no new or different type of equipment is being installed) and do not involve a change in the design, normal configuration, or basic operation of the plant. The proposed changes do not introduce any new accident initiators. The proposed changes do not involve significant changes in the fundamental methods governing normal plant operation and do not require unusual or uncommon operator actions. The proposed changes provide assurance that the plant will not be operated in a mode or condition that violates the assumptions or initial conditions in the safety analyses and that SSCs [structures, systems, and components] remain capable of performing their intended safety functions as assumed in the same analyses. Consequently, the response of the plant and the plant operator to postulated events will not be significantly different.

Therefore, the proposed TS change does 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.

Margin of safety is related to confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. The proposed changes address control rod scram test performance and acceptance criteria as well as control rod operability requirements. The scam test acceptance criteria and control rod operability restrictions are based on industry approved methodology and will continue to ensure control rod scram design functions and reactivity insertion assumptions used in safety analyses continue to be protected. The proposed changes also extend the frequency of testing control rod scram times while at-power from 120 days to 200 days. The proposed change ensures scram testing is performed and that test results verify acceptable operation of the control rods.

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 50.929(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Travis C. McCullough, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601.

Branch Chief: John P. Boska (Acting).

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

Date of amendment request: March 15, 2007.

Description of amendment request: The proposed amendment would revise containment systems surveillance requirements for Technical Specification (TS) 3/4.6.2, ''Depressurization, Cooling, and pH Control Systems.'' The proposed amendment would revise the frequency for ANO–2 TS Surveillance Requirement 4.6.2.1.d to require verification that spray nozzels are unobstructed following maintenance that could result in a nozzel blockage (loss of foreign material exclusion control) rather than performing an air or smoke flow test through each spray header every 5 years.

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

1. Do[es] the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The Containment Spray System (CSS) is not an initiator of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that may initiate an analyzed event. The proposed change will not alter the operation or otherwise increase the failure probability of any plant equipment that can initiate an analyzed accident. This change does not affect the plant design. There is no increase in the likelihood of formation of significant corrosion products. Due to their location at the top of the containment, introduction of foreign material into the spray headers is unlikely. Foreign materials exclusion controls during and following maintenance provides assurance that the nozzles remain unobstructed. Consequently, there is no significant increase in the probability of an accident previously evaluated.

The CSS is designed to address the consequences of a Loss of Coolant Accident (LOCA) or a Main Steam Line Break (MSLB). The Containment Spray System is capable of performing its function effectively with the single failure of any active component in the system, any of its subsystems, or any of its support systems. Therefore, the consequences of an accident previously evaluated are not significantly affected by the proposed change.

2. Do[es] the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will not physically alter the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation.

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

3. Do[es] the proposed change involve a significant reduction in a margin of safety?

Response: No.

The system is not susceptible to corrosion-induced obstruction or obstruction from sources external to the system. Strict controls are established to ensure the foreign material is not introduced into the CSS during maintenance or repairs. Maintenance activities that could introduce significant foreign material into the system require subsequent system cleanliness verification which would prevent nozzle blockage. The spray header nozzles are expected to remain unblocked and available in the event that the safety function is required. The capacity of the system would remain unaffected.

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: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Thomas G. Hiltz.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: March 1, 2007.

Description of amendment request: The proposed change would revise the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS) to add a note to the Required Actions of TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' Actions A.1 and B.1. GGNS TS 3.6.1.3 requires specific actions to be taken for inoperable PCIVs. The TS Required Actions include isolating the affected penetration by use of a closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. The new note would allow a relief valve to be used without being deactivated, to comply with TS 3.6.1.3, Actions A.1 and B.1, provided it has a relief setpoint of at least 1.5 times containment design pressure (*i.e.*, at least 23 pounds per square inch gauge) and meets one of the following criteria:

1. The relief valve is 1-inch nominal size or less, or

2. The flow path is into a closed system whose piping pressure rating exceeds the containment design pressure rating.

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.

Primary Containment Isolation Valves (PCIVs) are accident mitigating features designed to limit releases from the containment following an accident. The TS specify actions to be taken to preserve the containment isolation function if a PCIV is inoperable. These actions include isolating the penetration flow path by specific methods including, closed and de-activated automatic valves, closed manual valves, blind flanges, and check valves with flow through the valve secured. The current TS Actions do not specifically recognize a closed relief valve as an acceptable method of isolating a penetration flow path. Thus, special measures may need to be taken to comply with the TS Required Actions, such as replacing the relief valve with a blind flange or de-activating the relief valve by installing a gag. While such actions may provide additional assurance of preserving the containment isolation function, it may also have adverse safety affects such as disabling the overpressure protective safety feature, causing additional safety system unavailability time, and increasing occupational dose.

The proposed change would allow certain relief valves to be used for isolating the penetration flow path without being de-activated. The proposed TS changes do not alter the design, operation, or capability of PCIVs. Relief valves are designed to be normally closed to preserve the piping boundary integrity yet automatically open on an abnormal process pressure to protect the piping from overpressure conditions. Relief valves may also serve as passive containment isolation devices (i.e., they do not require mechanical movement to perform the isolation function). The proposed TS changes preserve both the containment isolation and piping overpressure protection functions.

The failure of a relief valve to remain closed during or following an accident is considered a low probability because relief valves are passive isolation devices that do not require mechanical movement to perform the isolation function and the relief setpoint provides sufficient margin to preclude the potential for premature opening due to containment post-accident pressures. Additional criteria are established to provide defense-in-depth protection. Relief valves that are one-inch or smaller provide an additional physical barrier in that, even in the unlikely event that a relief valve were to fail to remain fully closed during or following an accident, the size restriction would limit leakage such that a large early release would not occur. By definition, penetrations one-inch and smaller do not contribute to large early releases. Larger relief valves may be used as isolation devices provided that the containment penetration flow path through the relief valve would be contained in a closed system. In the unlikely event that a relief valve were to fail to remain closed, the leakage would be into a system which forms a closed loop outside primary containment and any containment leakage would return to primary containment through this closed loop.

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 proposed change does not introduce any new modes of plant operation or adversely affect the design function or operation of safety features. The proposed TS change allows use of existing plant equipment as compensatory measures to maintain the containment isolation design intent when the normal isolation features are inoperable. Since relief valves used for this purpose will not be disabled by gags or blind flanges, the system piping overpressure protection design feature will also be preserved.

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 safety margin associated with this change is that associated with preserving the containment integrity. NUREG–0800, the Standard Review Plan, recognizes that relief valves with relief setpoints greater than 1.5 times containment design pressure are acceptable as containment isolation devices. Closed relief valves with relief setpoints of this margin provide an isolation alternative that is less susceptible to a single failure (i.e., inadvertent opening) yet still preserves the overpressure protection that the component was intended to provide. The failure of a relief valve to remain closed during or following an accident is considered a low probability because relief valves are passive isolation devices that do not require mechanical movement to perform the isolation function and the relief setpoint provides sufficient margin to preclude the potential for premature opening due to containment post-accident pressures. Defense-in-depth containment leakage protection is provided by additional TS criteria that limit the use of relief valves to those one-inch or less in size or those where containment leakage would be into a closed system whose piping pressure rating exceeds the containment design pressure rating. Relief valves that are one-inch or smaller provide an additional physical barrier in that, even in the unlikely event that a relief valve were to fail to remain closed during or following an accident, the size restriction would limit leakage such that a large early release would not occur. In the unlikely event that a relief valve larger than one-inch were to fail to remain closed, the leakage would be into a system which forms a closed loop outside primary containment and any containment leakage would return to primary containment through this closed loop.

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: Terence A. Burke, Associate General Council—

Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: David Terao.

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

Date of amendment request: February 9, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 3.3.2, ''Engineered Safety Feature Actuation System Instrumentation,'' TS 3.5.2, ''Emergency Core Cooling System—Operating,'' TS 3.6.5, ''Containment Air Temperature,'' and TS 5.5.12, ''Containment Leakage Rate Testing Program.'' The revised TSs would be consistent with a proposed change to the Recirculation Spray System (RSS) pump start signal due to a modification to the containment sump screens.

The proposed amendment would also replace the use of LOCTIC with the Modular Accident Analysis Program-Design Basis Accident (MAAP–DBA) for calculating containment pressure, temperature, and condensation rates for input to the SWNAUA code. The calculation methodology change would ultimately change the aerosol removal coefficients used in dose consequence analysis.

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

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

Response: No. The proposed changes to the RSS pump start signal, the upper containment temperature technical specification (TS) limit, the peak containment internal pressure, the nomenclature for automatic switchover to the containment sump, and the containment sump screen visual inspection surveillance requirement do not involve any system or component that are accident initiators. The RSS is used for accident mitigation only. The Refueling Water Storage Tank (RWST) level and containment pressure instrumentation will continue to comply with all applicable regulatory requirements and design criteria (e.g., train separation, redundancy, single failure, etc.) following approval of the proposed changes. The design functions performed by the RSS and the containment are not changed by this license amendment request.

Delaying the start of the RSS pumps and the change to the upper containment temperature affect the long-term containment pressure and temperature profiles. The environmental qualification of safety-related equipment inside containment will be confirmed to be acceptable and accident mitigation systems will continue to operate within design temperatures and pressures. Delaying the RSS pump start reduces the emergency diesel generator loading in the early stage of a design basis accident and maintaining the staggered loading of the RSS pump starts avoids overloading on each emergency diesel generator at Unit 1. Staggered loading of the emergency diesel generator is not required for Unit 2.

The methodology change to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code will not involve a significant increase in the probability of an accident previously evaluated because this change in methodology does not impact accident initiators.

The loss of coolant accident (LOCA) has been evaluated using the guidance provided in Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.'' The radiological consequences of the remaining design basis accidents are not significantly impacted by the proposed changes. As demonstrated by the supporting analyses, the estimated dose consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room remain within the acceptance criteria of 10 CFR 50.67 as supplemented by Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1. In addition, the supporting analyses also demonstrates that the dose consequences in the Emergency Response Facility remain compliant with paragraph IV.E.8 of Appendix E, to 10 CFR part 50, Emergency Planning and Preparedness for Production and Utilization Facilities, regulatory guidance provided in Supplement 1 of NUREG–0737. The revised radiological analyses results in a slight increase in control room and off-site doses; however, the radiological analyses and evaluations developed in support of this application demonstrate that the proposed changes will not impact compliance with applicable regulatory requirements and will not involve a significant increase in the consequences of an accident previously evaluated. The slight increase in control room and off-site doses is more than offset by the increased assurance of adequate NPSH [net positive suction head] to the RSS pumps and Emergency Core Cooling System operability.

The safety analysis acceptance criteria will continue to be met following the proposed changes to the RSS pump start signal, visual sump inspection, TS containment upper temperature limit, peak containment internal pressure, nomenclature for automatic switchover to the containment sump and the change to the control room and off-site dose consequences analyses.

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. One of the proposed changes alters the RSS pump start circuitry by initiating the pump start from a coincident Containment Pressure High-High/[RWST] Level Low signal instead of from a timer. The RSS pump instrumentation will be included as part of the Engineered Safety Feature Actuation System (ESFAS) instrumentation in the TS and will be subject to the ESFAS surveillance requirements following approval of the proposed changes. The design of the RSS pump start instrumentation complies with all applicable regulatory requirements and design criteria. The failure modes have been analyzed to ensure that the revised RSS pump start circuitry can withstand a single active failure without affecting the RSS design functions. The RSS is an accident mitigation system only, so no new accident initiators are created.

It is not expected that the change in containment temperature will have a significant impact on equipment qualification. However, any equipment that must be replaced or re-qualified will be addressed prior to operation with the proposed change to RSS pump start. As a result any such equipment will not introduce new failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident.

The remaining changes do not change plant equipment design or function and therefore will not introduce new failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident.

Therefore, the proposed changes do 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 changes to the RSS pump start signal and the upper containment temperature limit affect the containment response and the LOCA dose analyses. Analyses demonstrate that containment design basis limits are satisfied and post-LOCA offsite and control room dose criteria will continue to be met following approval of the proposed changes.

The change to the containment sump visual inspection will not involve a significant reduction in a margin of safety because the revised surveillance will continue to provide adequate assurance the sump screens are not blocked with debris and that signs of corrosion will be detected.

The change to peak containment internal pressure will not result [in] a significant reduction in a margin of safety because the new pressure is lower for each of the units.

Although the control room and off-site doses slightly increase (due to a combination of the change to the start signal and the proposed methodology change), the increase will not involve a significant reduction in a margin of safety because operator and public exposure limits will continue to meet applicable regulatory requirements.

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: David W. Jenkins, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Acting Branch Chief: John P. Boska.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: March 8, 2007.

Description of amendment request: The proposed amendment would add Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.8 to allow a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The proposed change is consistent with TS Task Force (TSTF) change TSTF–372-A, Revision 4, ''Addition of LCO 3.0.8, Inoperability of Snubbers.''

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on November 24, 2004 (69 FR 68412). The licensee affirmed the applicability of the model in its application.

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 allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in Regulatory Guide 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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 Acting Branch Chief: John P. Boska.

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

Date of amendment request: February 15, 2007.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) Surveillance Requirement (SR) 3.8.4.2 to correct errors inadvertently introduced by Amendment No. 146. SR 3.8.4.2 currently requires that each battery charger be verified to supply greater than or equal to 150 amps for 250-volt DC subsystems, and greater than or equal to 50 amp for 125-volt DC subsystems. The licensee proposed to correct the errors by differentiating that the Division 1 battery chargers are verified to supply greater than or equal to 150 amps and the Division 2 battery chargers are verified to supply greater than or equal to 110 amps. The licensee stated that the Division 2 battery charger output current limiter is field-adjusted to supply 120 to 125 amps in order to stay within the electrical circuit breaker ratings in the associated distribution cabinet.

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 (NSHC). The NRC staff reviewed the licensee's analysis, and has performed its own analysis as follows:

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

No. The proposed amendment would only correct the battery chargers' DC supply current limits specified by SR 3.8.4.2. The current limits of the battery chargers were not considered to be a precursor to, and does not affect the probability of, an accident. In addition, there is no design or operation change associated with the proposed amendment. Therefore, the proposed amendment does not increase the probability of an accident previously evaluated.

The corrected DC supply current limits of the battery chargers will ensure that the batteries will be charged under as-designed conditions. The corrected limits will not decrease the functionality of the Division 1 or Division 2 battery chargers, or the functionality of the batteries the battery chargers support. Therefore, the plant systems required to mitigate accidents will remain capable of performing their design functions. As a result, the proposed amendment will not lead to a significant change in the consequences of any accident.

(2) Does the proposed amendment create the possibility of a new or different kind of

accident from any accident previously evaluated?

No. The proposed amendment does not involve a physical alteration of any system, structure, or component (SSC) or a change in the way any SSC is operated. The proposed amendment does not involve operation of any SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the revised acceptance value. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment would only change the current supply limits of the battery chargers. There will be no modification of any TSs limiting condition for operation, no change to any limit on previously analyzed accidents, no change to how previously analyzed accidents or transients would be mitigated, no change in any methodology used to evaluate consequences of accidents, and no change in any operating procedure or process. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Branch Chief: L. Raghavan.

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

Date of amendment requests: March 30, 2007.

Description of amendment requests: The proposed change will revise Technical Specifications (TSs) Surveillance Requirement (SR) 3.3.7.3.b, ''Loss of Voltage Function'' to a narrower voltage band and lower operating time for channel calibration testing, by replacing the undervoltage relays with the reset time significantly lower.

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 revises the Technical Specifications Surveillance Requirement 3.3.7.3.b allowable set point values of the Loss of Voltage Function for the channel calibration testing. This proposed change will allow Southern California Edison (SCE) to increase margin and conservatism for the loss of voltage relay settings and overall loop uncertainties while performing Loss of Voltage Signal (LOVS) channel calibration testing.

The loss of voltage function is detected by the LOVS relays installed on the 4.16 kV Safety Related busses. Normally, these devices are not considered to be accident initiators. The proposed change narrows the voltage operating band and lowers the allowable upper limit for this loss of voltage detection by use of the electronic type Basler BE1–27 under-voltage relays. However, the reset time of the relay [will be reduced] significantly. [Therefore, t]he proposed change does not impact 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 [an] accident previously evaluated?

Response: No.

The proposed allowable values for the LOVS relays voltage settings and the minimum operating voltage of the of[f]site power will provide acceptable level of protection for the plant equipment.

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

Response: No.

The proposed loss of voltage function is designed to ensure that plant equipment will not operate beyond its normal operating range for satisfactory operation of all the safety related equipment. The proposed loss of voltage function values will not affect the existing protection criterion for the plant equipment and will not reduce 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: Thomas G. Hiltz.

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 16, 2007.

Description of amendment request: The proposed amendment would permanently revise Technical Specification 2.2.1, Table 2.2–1, Functional Unit 17.A, Turbine Trip Low Trip System Pressure allowable value. The proposed revision was previously approved for one operating cycle at each unit.

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 revises the allowable value for reactor trip as a result of a turbine trip on low trip system pressure. This change will not alter any plant components, systems, or processes and will only provide a more appropriate value to assess operability of the associated pressure switches. Since the plant features and operating practices are not altered, the possibility of an accident is not affected. This reactor trip is not directly credited in SQN's [Sequoyah Nuclear Plant's] accident analysis and is maintained as an anticipatory trip to enhance the overall reliability of the reactor trip system. As such, there is not a specific safety limit associated with this function and the generation of a reactor trip based on low trip system pressure is above the required actuations to ensure acceptable mitigation of accidents. As the proposed change will continue to provide an acceptable anticipatory trip signal, the offsite dose potential is not affected by this change. 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.

As described above, this change will not alter any plant equipment or operating practices that have the ability to create a new potential for accident generation. The proposed change revises the operability limits for a function that generates a trip signal when appropriate conditions exist to require accident mitigation response. This type of function does not have the ability to create an accident as its purpose and function is to mitigate events. 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 proposed change will revise an allowable value for a reactor trip initiator that results from a turbine trip condition. This change will not alter the setpoint, and the calibration of the associated pressure switches will continue to be set at the current value. The allowable value change is in response to accuracy aspects of the instrumentation and does not alter the ability of this trip function to operate when and as needed to mitigate accident conditions. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Thomas H. Boyce.

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 51.22(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, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: January 19, 2007, as supplemented by letters dated March 13 and 22, 2007.

Brief description of amendment: The amendment modifies Technical Specifications 5.5.9 and 5.6.8 to add steam generator alternate repair criteria and additional steam generator reporting criteria at H. B. Robinson Steam Electric Plant, Unit No. 2.

Date of issuance: April 9, 2007.

Effective date: This license amendment is effective as of the date of issuance and shall be implemented within 60 days.

Amendment No.: 214.

Renewed Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register****:* January 30, 2007 (72 FR 4300). The March 13 and 22, 2007, supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 9, 2007.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: March 28, 2006, as supplemented by letters dated October 26, and December 4, 2006, and January 26, 2007.

Brief description of amendment: The amendment revises Millstone Power Station, Unit No. 3 Technical Specifications (TS) to delete redundant surveillance requirements pertaining to post-maintenance/post-modification testing.

Date of Issuance: March 29, 2007.

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

Amendment No.: 237.

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

Date of initial notice in ***Federal Register****:* May 23, 2006 (71 FR 29673). The supplemental 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.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2007.

No significant hazards consideration comments received: No.

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

Date of application of amendments: April 11, 2006, as supplemented October 24, 2006.

Brief description of amendments: The amendments revised the Technical Specifications requirements related to steam generator tube integrity consistent with the NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–449, ''Steam Generator Tube Integrity.'' These amendments also remove license conditions that become outdated with these TS changes. In addition, the amendments revised the organizational description in TS 5.2.1, which is solely administrative and unrelated to steam generator tube integrity.

Date of Issuance: April 2, 2007.

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

Amendment Nos.: 355, 357, 356.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register****:* January 3, 2007 (72 FR 149). The supplement dated October 24, 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 April 2, 2007. No significant hazards consideration comments received: No.

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

Date of application of amendments: April 11, 2006, as supplemented by letter dated March 14, 2007.

Brief description of amendments: The amendments added Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.8 to allow a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed with an approved Bases Control Program that is consistent with the TS Bases Control Program described in Section 5.5 of the applicable vendor's Standard Technical Specifications.

Date of Issuance: April 2, 2007.

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

Amendment Nos.: 356, 358, 357.

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

*Date of initial notice in **Federal Register**:* January 3, 2007 (72 FR 151). The supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on January 3, 2007 (72 FR 151). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 2, 2007.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois

Date of application for amendment: November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006, and February 15, February 23, and March 7, 2007.

Brief description of amendment: The amendments would revise the existing steam generator tube surveillance program using Technical Specification Task Force Traveler No. 449 (TSTF–449), Revision 4, ''Steam Generator Tube Integrity'' as a basis. The amendments would also revise TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' regarding the required SG inspection scope for Byron Station, Unit No. 2, during outage number 13 and subsequent operating cycle. A similar approval was granted for Braidwood Station, Unit 2 by letter from the NRC dated October 24, 2006.

Date of Issuance: March 30, 2007.

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

Amendment Nos.: 150/150, 144/144.

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

*Date of initial notice in **Federal Register**:* May 23, 2006 (71 FR 29676). The August 18 and September 28, 2006 and February 15, February 23, and March 7, 2007 supplements, contained clarifying information and did not change the staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 30, 2007.

No significant hazards consideration comments received: No.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: March 23, 2006, as supplemented by letters dated August 16 and November 28, 2006.

Description of amendment request: The amendment revises the Seabrook Station, Unit No. 1 Technical Specifications (TSs) Definitions, TS 3.4.5, ''Steam Generator (SG) Tube Integrity,'' and TS 3.4.6.2, ''Reactor Coolant System Operational Leakage'' consistent with Technical Specification Task Force (TSTF) Standard Technical Specification Traveler TSTF–449, ''Steam Generator Tube Integrity,'' Revision 4. Additionally the amendment creates TS 6.7.6.k. ''Steam Generator (SG) Program'' and TS 6.8.1.7, ''Steam Generator Tube Inspection Report,'' consistent with TSTF–449, Revision 4.

Date of Issuance: March 28, 2007.

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

Amendment No.: 115.

Facility Operating License No. NPF–86: The amendment revised the License and Technical Specifications.

*Date of initial notice in **Federal Register**:* April 25, 2006 (71 FR 23955). The licensee's August 16 and November 28, 2006, supplements provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the **Federal Register**, 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 28, 2007.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–266, Point Beach Nuclear Plant, Unit 1, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: July 11, 2006, as supplemented January 19, March 9 and 26, 2007.

Brief description of amendments: The amendment revises Technical Specification (TS) 5.5.8, ''Steam Generator Program,'' to change the inspection and repair criteria for portions of the tubes within the hot-leg region of the tubesheet for a single operating cycle. In addition, an administrative change corrects a page number in the TS Table of Contents and deletes two blank pages in TS Section 5.0.

Date of Issuance: April 4, 2007.

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

Amendment No.: 226.

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

*Date of initial notice in **Federal Register**:* August 29, 2006 (71 FR 51230). The supplements dated January 19, March 9 and 26, 2007, contained clarifying information and did not change the staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 2007.

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: December 20, 2006.

Brief description of amendment: The amendment removed annotations referencing Technical Data Book (TDB)–VIII, ''Equipment Operability Guidance,'' and annotations referencing Technical Specification Interpretations (TSIs) from the NRC Authority File of the Technical Specifications (TSs). These documents are used by Omaha Public Power District (OPPD) personnel for additional guidance in applying certain Limiting Conditions of Operation requirements to specific equipment and/or situations. OPPD has annotated references to these documents in the TS copies used at the Fort Calhoun Station, Unit No.1 (FCS); however, these annotations were inadvertently included into the NRC Authority File and are not officially part of the FCS TS. The amendment also corrected a discrepancy in TS 2.10.4(1)(c).

Date of Issuance: April 3, 2007.

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

Amendment No.: 249.

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

*Date of initial notice in **Federal Register**:* January 30, 2007 (72 FR 4308).

The Commission's related evaluation of the amendment is contained in a safety evaluation dated April 3, 2007.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: April 6, 2006, as supplemented by letters dated January 19, and February 27, 2007.

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

Date of Issuance: March 29, 2007.

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

Amendment No.: 262.

Facility Operating License No. DPR–75: The amendment revised the TSs and the License.

*Date of initial notice in **Federal Register***: July 18, 2006 (71 FR 40753). The letters dated January 19, and February 27, 2007, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 2, 2006, as supplemented by letter dated October 19, 2006.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.8.1, ''AC [alternating current] Sources—Operating,'' and TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' to increase the required amount of stored diesel fuel oil to support a change to Ultra Low Sulfur Diesel fuel from California diesel fuel presently in use. This change in the type of fuel oil is mandated by California air pollution control regulations.

Date of Issuance: April 4, 2007.

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

Amendment Nos.: Unit 2—211; Unit 3—203.

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

Date of initial notice in ***Federal Register:*** July 18, 2006 (71 FR 40754). The supplemental letter dated October 19, 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 no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 2007.

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: February 21, 2006.

Brief description of amendments: The amendments revised Technical Specifications 1.1, ''Definitions,'' and 3.4.16, ''RCS [Reactor Coolant System] Specific Activity.'' The revisions replaced the current Limiting Condition for Operation (LCO) 3.4.16 limit on RCS grossspecific activity with limits on RCS Dose Equivalent I–131 (DEI) and Dose Equivalent Xe-133 (DEX). The conditions and required actions for LCO 3.4.16 not being met, and surveillance requirements for LCO 3.4.16, are revised. The modes of applicability for LCO 3.4.16 are extended. TS Figure 3.4.16–1 on the limit for DEI with respect to rated thermal power is deleted.

Date of issuance: March 29, 2007.

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

Amendment Nos.: 137/137.

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

*Date of initial notice in **Federal Register***: February 27, 2007 (72 FR 8805).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 29, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 17, 2006.

Brief description of amendment: The amendment revised Technical Specifications (TSs) 2.1.1, ''Reactor Core SLs [Safety Limits],'' 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' 3.4.1, RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,'' and 5.6.5, ''Core Operating Limits Report (COLR).'' The changes (1) relocated certain operating cycle-specific core operating limits, including TS Figure 2.1.1–1, ''Reactor Core Safety Limits,'' from the TSs to the plant COLR, (2) added two new safety limits for departure from nucleate boiling ratio and peak fuel centerline temperature, and (3) added topical reports to TS 5.6.5 and had the reports cited by only the report title and number. TS 5.6.5 was expanded to include the limits from TSs 2.1.1, 3.3.1, and 3.4.1.

Date of Issuance: April 2, 2007.

Effective date: As of its date of issuance and shall be implemented within 90 days from the date of issuance. The final TS Bases changes including the licensee's application dated August 17, 2006, will be processed under the licensee's program for updates to the TS Bases, in accordance with TS 5.5.14, at the time this amendment is implemented. The final changes to the COLR including those in the licensee's application dated August 17, 2006, will be submitted to the NRC in accordance with the update process covered by TS 5.6.5.d.

Amendment No.: 183.

Facility Operating License No. NPF–30: The amendment revised the

Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** January 16, 2007 (72 FR 1781).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 2, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 26, 2006, as supplemented on January 19, 2007.

Brief Description of amendments: These amendments revised the Technical Specification (TS) requirements related to steam generator tube integrity and Reactor Coolant System leakage definitions and requirements. The TSs were revised to implement TS Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator Tube Integrity,'' (TSTF–449, Rev. 4) with minor deviations to be consistent with Surry's custom TSs.

Date of Issuance: March 29, 2007.

Effective date: As of date of issuance and shall be implemented within 180 days.

Amendment Nos.: 251, 250.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments changed the licenses and the technical specifications.

Date of initial notice in ***Federal Register***: August 15, 2006 (71 FR 46941). The supplement dated January 19, 2007, 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 29, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 5, 2006, as supplemented on September 21 and November 20, 2006.

Brief Description of amendments: These amendments revised the main control room (MCR) and emergency switchgear room (ESGR) air-conditioning system (ACS) Technical Specifications to reflect the completion of permanent modifications to the equipment and associated power supply configuration. The revisions include the addition of requirements and/or action statements addressing the inoperability of two or more air handling units (AHUs) on a unit, as well as AHU powered from an H emergency bus. The proposed change, paralleling requirements in the Improved Technical Specifications, also adds MCR and ESGR ACS requirements during refueling operations and irradiated fuel movement in the fuel building. In addition, the proposed change clarified the service water requirements for the ACS chillers that serve the MCR and ESGRs.

Date of Issuance: April 2, 2007.

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

Amendment Nos.: 252, 251.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments changed the licenses and the technical specifications.

Date of initial notice in ***Federal Register***: September 26, 2006 (71 FR 56193). The supplements dated September 21 and November 20, 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 April 2, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 16th day of April 2007.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E7–7534 Filed 4–23–07; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Notice of Cancellation of Public Hearing on Potential Withdrawal of Tariff Concessions and Increase in Applied Duties in Response to European Union (EU) Enlargement

AGENCY: Office of the United States Trade Representative.

ACTION: Notice of cancellation of April 24, 2007 public hearing concerning a list of goods for which tariff concessions may be withdrawn and duties may be increased in the event the United States cannot reach agreement with the European Communities (EC) for adequate compensation owed under World Trade Organization (WTO) rules as a result of EU enlargement.

SUMMARY: On March 22, 2007, USTR published FR Doc E7–5268 (Vol. 72, No. 55) announcing that the Trade Policy Staff Committee (TPSC) was seeking public comment on a list of goods for which U.S. tariff concessions may be withdrawn and applied duties may be raised and announcing that the TPSC will hold a public hearing on Tuesday, April 24, 2007, on the list. All respondents to this notice have chosen to submit their comments in writing only and there were no requests to testify. Therefore, the April 24 public hearing will be cancelled.

The United States is continuing to negotiate with the EU regarding the EU's provision of adequate and permanent compensation to the United States for an event that increased duties on U.S. imports to EU markets above WTO bound rates of duty. On January 1, 2007, as part of its enlargement process, the EU raised tariffs above bound rates on some imports into the countries of Romania and Bulgaria. If this issue is not resolved, the United States may seek to exercise its rights under Article XXVIII of the *General Agreement on Tariffs and Trade 1994* (''GATT 1994'') to withdraw substantially equivalent concessions and raise tariffs on select goods primarily supplied by the EU.

FOR FURTHER INFORMATION CONTACT: Questions should be directed to: Laurie Molnar, Director for European Trade Issues, (202) 395–3320; Office of the United States Trade Representative.

Carmen Suro-Bredie,

Chairman, Trade Policy Staff Committee.

[FR Doc. E7–7809 Filed 4–23–07; 8:45 am]

BILLING CODE 3190–W7–P

POSTAL SERVICE

Philadelphia, PA 30th Street Post Office Property Disposition

AGENCY: Postal Service.

ACTION: Notice.

SUMMARY: Notice is hereby given of the disposition of Postal Service(tm) property, the 30th Street Main Post Office located in Philadelphia, PA.

DATES: Comments must be submitted on or before April 30, 2007.

ADDRESSES: Comments may be mailed to Dallan Wordekemper, Postal Service, Federal Preservation Officer, 4301