NUCLEAR REGULATORY COMMISSION

Southern Nuclear Operating Company; Notice of Receipt and Availability of an Application for an Early Site Permit for the Vogtle ESP Site

On August 15, 2006, the Nuclear Regulatory Commission (NRC, the Commission) received an application from Southern Nuclear Operating Company filed pursuant to Section 103 of the Atomic Energy Act and 10 CFR part 52, for an early site permit (ESP) for a location in eastern Georgia (near Waynesboro, Georgia) identified as the Vogtle ESP site.

An applicant may seek an ESP in accordance with Subpart A of 10 CFR part 52 separate from the filing of an application for a construction permit (CP) or combined license (COL) for a nuclear power facility. The ESP process allows resolution of issues relating to siting. At any time during the period of an ESP (up to 20 years), the permit holder may reference the permit in an application for a CP or COL.

Subsequent **Federal Register** notices will address the acceptability of the tendered ESP application for docketing and provisions for participation of the public and other parties in the ESP review process.

A copy of the application is available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland and via the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* The accession number for the application is ML062290246.

Future publicly available documents related to the application will also be posted in ADAMS. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC Public Document Room staff by telephone at 1–800–397–4209 or 301–415–4737, or by e-mail to *pdr@nrc.gov.* The application is also available to local residents at the Burke County Library, in Waynesboro, Georgia, and it will be available on the NRC Web page at *http://www.nrc.gov/reactors/new-licensing/esp.html.*

Dated at Rockville, Maryland, this 21st day of August, 2006.

For the Nuclear Regulatory Commission.

David B. Matthews,

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

[FR Doc. E6–14285 Filed 8–28–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Notice

DATE: Weeks of August 28, September 4, 11, 18, 25, October 2, 2006.

PLACE: Commissioners' Conference Room, 1155 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of August 28, 2006

There are no meetings scheduled for the week of August 28, 2006.

Week of September 4, 2006—Tentative

Wednesday, September 6, 2006

1:50 p.m. Affirmation Session (Public) (Tentative)
- a. Pacific Gas & Elec. Co. (Diablo Canyon ISFSI), Docket No. 72–26–ISFSI ''Motion by San Luis Obispo Mothers for Peace, Sierra Club, and Peg Pinard for Declaratory and Injunctive Relief with respect to Diablo Canyon ISFSI''. (Tentative).
- b. AmerGen Energy Company, LLC (License Renewal for Oyster Creek Nuclear Generating Station) Docket No. 50–0219, Legal challenges to LBP–06–07 and LBP–06–11. (Tentative).
- c. Pa'ina Hawaii, LLC, LBP–06–4, 63 NRC 99 (2006) and LBP–06–12, 63 NRC 409 (2006). (Tentative).

Week of September 11, 2006—Tentative

Monday, September 11, 2006

9:30 a.m. Discussion of Security Issues (Closed—Ex. 1).

1:30 p.m. Discussion of Security Issues (Closed—Ex. 1& 3).

Tuesday, September 12, 2006

9:30 a.m. Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: Shawn Smith, 301–415–2620).

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

1 p.m. Discussion of Security Issues (Closed—Ex. 1).

Week of September 18, 2006—Tentative

There are no meetings scheduled for the week of September 18, 2006.

Week of September 25, 2006—Tentative

There are no meetings scheduled for the week of September 25, 2006.

Week of October 2, 2006—Tentative

There are no meetings scheduled for the week of October 2, 2006.

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*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Michelle Schroll, (301) 415–1661.

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

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

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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: August 24, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–7236 Filed 8–25–06; 9:49 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory

Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 4, 2006 to August 17, 2006. The last biweekly notice was published on August 15, 2006 (71 FR 46929).

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

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene.

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

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

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner/requestor to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

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

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: April 11, 2006.

Description of amendment request: The proposed amendment would modify Technical Specification 5.6.5 ''Core Operating Limits Report (COLR)'' to add two U.S. Nuclear Regulatory Commission-approved topical reports to the COLR methodologies list. These topical reports allow the use of S–RELAP5 thermal-hydraulic analysis code for accident safety analyses.

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

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

The two topical reports have been reviewed and approved by the NRC for use in determining core operating limits. The core operating limits to be developed using the new methodologies for HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2, will be established in accordance with the applicable limitations as documented in the NRC Safety Evaluation Reports. In a May 11, 2001, NRC Safety Evaluation Report, the NRC concluded that the S–RELAP5 code is capable of addressing the thermal-hydraulic response of the target non-LOCA [loss-of-coolant accident] events in a conservative manner and is, therefore, an acceptable replacement for the ANF–RELAP code. In the May 19, 2004, Safety Evaluation Report for Revision 1 to EMF–2310(P)(A), the NRC concluded that the code remained acceptable for use for the non-LOCA events. In a March 15, 2001, Safety Evaluation Report, the NRC concluded that the code was acceptable for use for small break LOCA analyses at Westinghouse pressurized water reactors.

The proposed change, by itself, does not impact the current design bases. The proposed change enables the use of new methodologies to re-analyze certain events. Revised analyses may either result in continued conformance with design bases, or may change the design bases. If design basis changes result from a revised analysis, then the specific design changes will be evaluated in accordance with HBRSEP, Unit No. 2, design change procedures and 10 CFR 50.59.

The proposed change does not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fission product barriers during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed methodologies will ensure that the plant continues to meet applicable design and safety analyses acceptance criteria. The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are impacted and there are no adverse effects on the factors that contribute to offsite or onsite dose as a result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, and components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed accident.

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

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

The proposed change does not involve any physical alteration of plant systems, structures, or components, other than allowing for fuel design in accordance with NRC-approved methodologies. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no change to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There is no impact on any margin of safety resulting from the incorporation of these new topical reports into the Technical

Specifications. If design basis changes result from a revised analysis that uses these new methodologies, the specific design changes will be evaluated in accordance with HBRSEP, Unit No. 2, design change procedures and 10 CFR 50.59. Any potential reduction in the margin of safety would be evaluated for that specific design change.

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

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

Attorney for licensee: 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: L. Raghavan.

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: July 17, 2006.

Description of amendment request: The proposed amendment would revise the containment pressure requirements specified in Surveillance Requirements 3.6.8 and 5.5.16 due to a revision in the Loss-of-Coolant Accident (LOCA) containment pressure 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?

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The revised post-LOCA containment pressure and temperature analysis used more conservative assumptions and the increase in the calculated peak pressure was approximately 1 psig. The revised value of 41.49 psig remains less than the containment design pressure of 42 psig. The increase in the calculated peak temperature was approximately 2 °F, which was analyzed to have no impact on structures or equipment. Although there is an increase in the calculated pressure, the allowable containment leakage rate, as measured at the peak pressure, is not being changed. Since there is no increase in the allowable leakage, there is no increase in consequences. The proposed change is related to containment pressure analysis. There are no physical changes being made to the plant, or to the manner in which the plant is operated. Surveillance procedures for containment leakage have been conservatively testing at pressures in excess of 42 psig and surveillance procedures for the Isolation Valve Seal Water System have been conservatively testing at pressures in excess of 46.2 psig. The change can have no impact on the probability of an accident occurring. Therefore, the proposed change does 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 previously evaluated?

No. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. There are no physical changes being made to the plant or to the manner in which the plant is operated. Surveillance procedures for containment leakage have been conservatively testing at pressures in excess of 42 psig and surveillance procedures for the Isolation Valve Seal Water System have been conservatively testing at pressures in excess of 46.2 psig. The revised containment analysis results in a calculated peak containment pressure that remains less than the containment design pressure. The increase in the calculated peak temperature was analyzed to have no impact on structures or equipment. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change does not involve a significant reduction in the margin of safety. The proposed change imposes more conservative surveillance test requirements. The calculated increase in post-LOCA peak containment pressure is only 1 psig and the revised value of 41.49 psig remains less than the containment design pressure of 42 psig. The increase in the calculated peak temperature was approximately 2 °F, which was analyzed to have no impact on structures or equipment. Although there was an increase in the calculated pressure, the allowable containment leakage rate, as measured at the peak pressure, is not being changed. Therefore, this change does not involve a significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: July 12, 2006.

Description of amendment request: The proposed amendment would modify Conditions, Required Actions and Completion Times associated with the inoperability of one or more emergency diesel generators (EDGs) in Technical Specification (TS) 3.8.1.

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

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

This license amendment request proposes a change to extend the Technical Specification 3.8.1, ''AC Sources-Operating,'' Completion Time. This change allows a single EDG to be inoperable for 7 days more than Technical Specification 3.8.1 currently provides. The Required Actions for CTG [Combustion Turbine Generator] 11–1 are also removed from Condition A and TSTF [Technical Specification Task Force] -439 is implemented for TS 3.8.1, removing the second Completion Times.

The EDGs are safety related components which provide backup electrical power supply to the onsite ESF [Engineered Safety Feature] power distribution system. CTG 11–1 provides backup electrical power to the Division 1 power distribution system. Neither the EDGs nor CTG 11–1 are accident initiators, thus these changes do not increase the probability of a previously evaluated accident.

The plant ESF power distribution systems consist of two divisions for 100% redundancy. Accident analyses demonstrate that only one division is required for accident mitigation. Thus, with one division inoperable the other division is capable of performing the required safety function. Design basis analyses are not required to be performed assuming extended loss of all power supplies to the plant ESF power distribution system. Thus, this change does not involve a significant increase in the consequences of a previously analyzed accident.

The proposed change also eliminates the second Completion Time from TS 3.8.1. These second Completion Times are not an initiator to any accident previously evaluated. As a result, the probability of an accident previously evaluated is not affected. The consequences of an accident during the revised Completion Times are no different than the consequences of the same accident during the existing Completion Times. As a result, the consequences of an accident previously evaluated are 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. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The changes do not alter any assumptions made in the safety analysis.

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

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

This license amendment request proposes Technical Specification changes to extend the Technical Specification 3.8.1, ''AC Sources-Operating,'' Completion Time for an inoperable EDG to 14 days. These changes allow an emergency diesel generator to be inoperable for 7 days more than TS 3.8.1 currently provides.

Deterministic and probabilistic risk assessments evaluated the effect of the proposed TS changes on the availability of an electrical power supply to the plant emergency safeguards features systems. These assessments concluded that the proposed TS changes do not involve a significant increase in the risk of power supply unavailability.

This license amendment request proposes TS changes to remove the Required Actions for CTG 11–1 from TS 3.8.1 Condition A. If CTG 11–1 is inoperable at the same time that any single EDG is inoperable for the entire proposed 14 day period with no other equipment in maintenance, the risk remains within RG [Regulatory Guide] 1.174 (Reference 2 [in the application]) thresholds for a ''very small'' classification.

The proposed change to delete the second Completion Time does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed changes will not result in plant operation in a configuration outside of the design basis.

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

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

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

NRC Acting Branch Chief: Martin Murphy.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Power Station, Unit No. 2, (MPS2) New London County, Connecticut

Date of amendment request: June 13, 2006.

Description of amendment request: The proposed amendment would allow changes to the Technical Specifications based on the radiological dose analysis margins obtained by using an alternate source term consistent with 10 CFR 50.67.

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

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

No.

The proposed amendment does not involve a significant increase in the probability or consequence of an accident previously analyzed. The MPS2 control room emergency ventilation system only functions following the initiation of a design basis radiological accident. Therefore, the change to the value used for in-leakage rate test acceptance criteria following a design basis accident will not increase the probability of any previously analyzed accident. The proposed 200 cfm control room habitability envelope inleakage surveillance acceptance criteria has no adverse impact on control room habitability analyses for postulated toxic chemical release events. These habitability analyses do not credit automatic or manual isolation of the control room fresh air ventilation flow during a toxic chemical release event. The control room's forced ventilation fresh air exchange rate (*e.g.*, 800 cfm) is much greater than the proposed 200 cfm envelope inleakage rate acceptance criteria. The MPS2 containment purge valve isolation signal is not credited in the accident analyses. The requirements contained in this specification do not meet any of 10 CFR 50.36(c)(2)(ii) criteria on items for which technical specifications must be established. Deletion of this technical specification will not increase the probability of any previously analyzed accident. The MPS2 containment and the containment systems function to prevent or control the release of radioactive fission products following a postulated accident. Therefore, the change to the value used for primary to secondary leak rate acceptance criteria, and for all secondary containment bypass leakage paths following a design basis accident, will not increase the probability of any previously analyzed accident.

These systems are not initiators of any design bases accident. Revised dose calculations, which take into account the changes proposed by this amendment and the use of the alternative source term, have been performed for the MPS2 design basis radiological accidents. The results of these revised calculations indicate that public and control room doses will not exceed the limits specified in 10 CFR 50.67 and Regulatory Guide 1.183. There is not a significant increase in predicted dose consequences for any of the analyzed accidents. Therefore, the proposed changes do not involve a significant increase in the consequences of any previously analyzed accident.

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

No.

The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the UFSAR. Although the proposed changes could affect the operation of the control room emergency ventilation system, and the containment and containment systems following a design basis radiological accident, none of these changes can initiate a new or different kind of accident since they are only related to system capabilities that provide protection from accidents that have already occurred. These changes do not alter the nature of events postulated in the UFSAR nor do they introduce any unique precursor mechanisms. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from those previously analyzed.

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

No.

The implementation of the proposed changes does not reduce the margin of safety. The proposed changes for the control room ventilation system, and the containment and containment systems do not affect the ability of these systems to perform their intended safety functions to maintain dose less than the required limits during design basis radiological events. The radiological analysis results, when compared with the revised TEDE [total effective dose equivalent] acceptance criteria, meet the applicable limits. These acceptance criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting the stated limits demonstrates adequate protection of public health and safety. It is thus concluded that the margin of safety will not be reduced by the implementation of the changes.

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 Acting Branch Chief: Brooke D. Poole.

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

Date of amendment request: July 19, 2006.

Description of amendment request: The proposed amendment will revise reactor core safety limits Technical Specifications (TSs) and relocate the reactor core safety limit figure to the Core Operating Limits Report (COLR) in the Millstone Power Station, Unit 3, Technical Requirements Manual.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, 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 relocation of cycle-specific core operating limits from the TS to the COLR and the addition of the RCS [reactor coolant system] total flow rate to the COLR has no influence or impact on the probability or consequences of a design basis accident. Adherence to the COLR and methodologies acceptable for establishing COLR parameters continues to be controlled by TS. The proposed amendment still requires exactly the same actions to be taken when or if limits are exceeded. Each accident analysis addressed in the final safety analysis report (FSAR) will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluation of new core designs are bounded by previously accepted analyses. This examination, which will be performed in accordance with the requirements of 10 CFR 50.59, ensures that future core designs will not involve a significant increase in the probability or consequences 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.

The relocation and addition of the cycle-specific variables to the COLR and adding new document references in TS Section 6.9.1.6.b does not influence or impact, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operations will be altered as a result of these proposed changes. The cycle-specific variables are calculated using NRC-approved methods and submitted to the NRC to allow the staff to continue to trend the values of these limits. The TS will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded. Therefore the proposed amendment does not in any way 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 changes have no impact on plant equipment operation. The proposed changes do not revise any setpoints or acceptance criteria assumed in the accident analyses. Therefore, the proposed changes will not result in 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Acting Branch Chief: Brooke D. Poole.

Entergy Nuclear Operations, Inc., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2 (IP2), Westchester County, New York

Date of amendment request: July 10, 2006.

Description of amendment request: Energy Nuclear Operations, Inc., is planning to operate an Independent Spent Fuel Storage Installation (ISFSI) facility at IP2 using the HOLTEC HI-STORM 100 Cask System. To support this activity, the proposed amendment adds Spent Fuel Cask loading requirements to IP2 Technical Specifications (TSs).

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 will revise the Indian Point Unit 2 TSs associated with the SFP [spent fuel pool] to assure that the regulatory requirements related to criticality in the SFP and applied to the Holtec HI-STORM 100 Multi-Purpose Canister MPC–32 when in the SFP are reflected in the IP2 TS. The proposed change does not require any physical changes to Part 50 structures, systems, or components, nor will their performance requirements be altered. Therefore, the response of the plant to previously analyzed accidents and related radiological releases will not be adversely impacted, and will bound those postulated during cask loading activities in the cask storage area.

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.

Existing fuel handling procedures and associated administrative controls remain applicable for cask loading operations within the SFP. Additionally, the soluble boron concentration required to maintain $k_{eff} \leq 0.95$ for postulated criticality accidents associated with cask loading operations was also evaluated. The results of the analyses, using a methodology previously approved by the NRC [Nuclear Regulatory Commission], demonstrate that the amount of soluble boron required to compensate for the positive reactivity associated with these postulated accidents (371 ppm [parts per million]) remains well below the existing spent fuel pit minimum boron concentration limit of 2000 ppm. Accordingly, the same limit has been proposed for cask loading operations in the cask storage area.

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

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

Response: No.

An NRC approved methodology was used to perform the criticality analysis which provides the basis to incorporate a new family of burnup versus enrichment curves, for various cooling times, into the plant Technical Specifications to ensure criticality requirements are met during spent fuel cask loading. Accordingly, the existing minimum boron concentration limit for the spent fuel pit of 2000 ppm will continue to remain bounding during cask loading operations. This determination accounts for uncertainties at a 95 percent probability, 95 percent confidence level. Should it be postulated that a boron dilution event does occur during this time period, k_{eff} will remain less than 1.0 should the cask storage area become fully flooded with unborated water.

Therefore, there will not be a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Richard J. Laufer.

Exelon Generation Company, LLC, Docket No. 50–249, Dresden Nuclear Power Station (DNPS), Unit 3, Grundy County, Illinois

Date of amendment request: July 21, 2006.

Description of amendment request: The proposed amendment would revise the values of the safety limit minimum critical power ratio (SLMCPR) in Technical Specification Section 2.1.1, ''Reactor Core SLs [Safety Limits].'' Specifically, the proposed change would require that for Unit 3, the minimum critical power ratio (MCPR) for Global Nuclear Fuel fuel shall be ≥ 1.10 for two recirculation loop operation, or ≥ 1.11 for single recirculation loop operation. Additionally, the proposed change would require that MCPR for Westinghouse fuel shall be ≥ 1.12 for two recirculation loop operation, or ≥ 1.14 for single recirculation loop operation.

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 probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC [Nuclear Regulatory Commission]-approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the SLMCPR for DNPS, Unit 3, Cycle 20 such that the fuel is protected during normal operation and during plant transients or anticipated operational occurrences (AOOs).

Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during plant transients or AOOs. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (*i.e.*, that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs) is met. Since the proposed change does not affect operability of plant systems designed to mitigate any consequences of accidents, the consequences of an accident previously evaluated are not expected to increase.

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.

Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change does not involve any plant configuration modifications or changes to allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for DNPS, Unit 3, Cycle 20.

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 SLMCPR provides a margin of safety by ensuring that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed change will ensure the current level of fuel protection is maintained by continuing to ensure that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed SLMCPR values were developed using NRC-approved methods. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (*i.e.*, that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated) is met.

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

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

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

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: June 6, 2006.

Description of amendment request: The proposed amendment would revise the Ventilation Filter Test Program (VFTP) in Technical Specification (TS) 5.5.7. The license amendment is a corrective action to revise the flow rate units specified in the VFTP from standard cubic feet per minute to cubic feet per minute. This amendment will ensure the PNPP TS are consistent with plant design documentation, testing criteria, and the industry.

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:

The ESF [Engineered Safety Feature] Ventilation systems reduce the concentration of airborne radioactive contaminants following a design basis accident and therefore are not initiators of design bases accidents. The proposed amendment does not change the manner in which the ESF ventilation systems are operated or tested. Implementation of the proposed amendment will ensure the ESF ventilation systems perform their function when called upon and does not affect the plant operations, design function or analysis that verifies the capability of a [plant] structures, systems or components.

The proposed amendment does not affect the design of the ESF ventilation systems, the operational characteristics of the ESF ventilation systems, the interfaces between the ESF ventilation systems and those plant systems they support, or the reliability of the ESF ventilation systems.

Therefore, the ESF ventilation systems will be capable of performing their accident mitigation function and there is no increase in the probability or consequences of an accident already evaluated.

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

Response:

The proposed amendment introduces no new mode of plant operation and does not involve a physical modification to the plant. New equipment is not installed with the proposed amendment, nor does the proposed amendment cause existing equipment to be operated in a new or different manner.

Since the proposed changes do not involve a change to the plant design or operation, no new system interactions are created by this change. The proposed amendment does not produce any parameters or conditions that could contribute to the initiation of accidents different from those already evaluated in the Updated Safety Analysis Report.

The changes to the VFTP do not affect the assumed accident performance of the ESF Ventilation systems, nor [*sic*] any plant structure, system or component previously evaluated.

Therefore, 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?

Response:

The proposed amendment does not impact the ESF ventilation systems performance, including the capability for each ESF ventilation system to attain and maintain required air flow assumed in the plant safety analysis.

The proposed amendment does not involve a significant reduction in a margin of safety since the operability of the ESF ventilations [sic] systems continues to be determined as required to support the capability of the ESF ventilations [sic] systems to provide the required ventilation, filtration and temperature control to mitigate the consequences of an accident.

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

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

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: May 25, 2006.

Description of amendment request: The proposed license amendment revises the requirements in the Crystal River Unit 3 Improved Technical Specification related to steam generator tube integrity. The licensee states that the changes are consistent with NRC-approved Technical Specification (TS) Task Force (TSTF) Change Traveler, TSTF–449, ''Steam Generator Tube Integrity,'' Revision 4. The availability of this technical specification improvement was announced in the **Federal Register** on May 6, 2005, as part of the consolidated line item improvement process.

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

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

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

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

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

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

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT 1–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT 1–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT 1–131 are at the TS values before the accident.

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

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

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

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

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

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

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

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

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

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

The NRC staff proposes to determine that the 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: L. Raghavan.

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

Date of amendment request: July 11, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.8, ''Steam Generator (SG) Program,'' to exclude the portion of the tube below 17 inches from the top of the tubesheet from the SG tube inspection requirements for Unit 1 on a one-time basis for a single operating cycle. In addition, administrative changes are proposed to correct a page number in the TS table of contents and delete two blank pages in TS Section 5.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change revises Technical Specification (TS) 5.5.8, ''Steam Generator (SG) Program'' to redefine the PBNP Unit 1 primary pressure boundary for purposes of the SG tube inspection requirements on a one-time basis for Unit 1 Refueling Outage 30 and the subsequent operating cycle. The redefined primary pressure boundary is relocated from the seal weld at the bottom of the SG tube to the tube-to-tubesheet mechanical interface. The required structural integrity margins of the SG tubes in this area are unaffected by this change and will be maintained by the SG tubesheet. SG tubes are hydraulically expanded into the tubesheet. Steam generator tube rupture is constrained by the tubesheet for tubes with cracks in the tubesheet. This constraint results from the hydraulic expansion process which restricts further expansion of the tube, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Thermal expansion and differential pressure also restrain the tube axially. For conservatism, hydraulic preload was not factored into the analysis.

The proposed change continues to require that the SG Program include performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification).

The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. The analysis shows that structural integrity retains acceptable safety factors against burst under normal steady state full power operation primary-to-secondary pressure differential and against burst applied to the design basis accident primary-to-secondary pressure differentials. The analysis also shows that accident induced leakage is bound by twice the normal operating leakage and well below the accident analysis assumption for each stream generator. The primary to secondary operational LEAKAGE limit is not changed.

The planned inspection and supporting analysis provide reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout the operating cycle and in the unlikely event of a design basis accident. The proposed change does not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. The plant technical specification limits for operational LEAKAGE and for DOSE EQUIVALENT I–131 in primary coolant, which ensure the plant is operated within its analyzed condition, are unaffected by the proposed change. Therefore, the proposed change does not significantly increase the consequences of any accident previously evaluated.

The proposed change does not significantly affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature actuation setpoints, accident mitigation capabilities, or accident analysis assumptions or inputs.

Therefore, the probability or consequences of any accident previously evaluated will not be significantly increased as a result of the proposed change.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed change will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. Primary to secondary leakage that may be experienced during all plant conditions will continue to be monitored to ensure it remains within current accident analysis assumptions. The proposed change does not affect the method of operation of the SGs, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. Equipment important to safety will continue to operate as designed. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in adverse conditions or result in any increase in the challenges to safety systems. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The steam generators (SGs) are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. They are also relied upon to remove residual heat from the primary system. The safety function of an SG is maintained by ensuring the integrity of its tubes. Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change redefines the PBNP Unit 1 primary pressure boundary from the tube end weld to 17 inches below the top of the tubesheet and incorporates revisions to the inspection criteria for SG tube inspection in the tubesheet. The SG operating environment is not affected by the change. The proposed change maintains the required structural margins of the SG tubes for both normal and accident conditions.

For cracking located within the tubesheet, steam generator tube rupture is constrained by the tubesheet. For circumferentially oriented cracking, the associated analysis for the proposed change validates that 17 inches of degradation free expanded tubing provides the necessary resistance to tube pullout with applicable safety factors applied.

The revised inspection criteria continue to verify SG tube integrity. The safety function of the affected components will be maintained with the redefined primary pressure boundary.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems or components (SSCs) important to safety. Therefore, the requested change will not result in a significant reduction in the margin of safety.

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

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

NRC Acting Branch Chief: Martin Murphy.

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

Date of amendment request: October 13, 2005.

Description of amendment request: The proposed amendment would modify the licensing bases of SSES 1 and 2 by adopting the Alternative Source Term (AST) methodology which replaces the current accident source term with an AST. The AST is characterized by the composition and magnitude of the radioactive material, the chemical and physical form of the radionuclides, and the timing of the releases of these radionuclides. The exceptions would be that the current Technical Information Document (TID)14844 accident source term would remain the licensing basis for (1) equipment qualification, (2) NUREG–0737 evaluations other than Control Room Habitability Envelope (CRHE) doses, and (3) Final Safety Analysis Report (FSAR) accidents not included in Regulatory Guide 1.183.

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 of occurrence or consequences of an accident previously evaluated?

Response: No.

Adoption of the AST and pursuant TS [Technical Specification] changes, changes to the TS's to address NRC Generic Letter 2003–01 (Reference 12.1) [see application dated October 13, 2005] and the changes to the atmospheric dispersion factors, have no impact to the initiation of DBAs [design basis accidents]. Once the occurrence of an accident has been postulated, the new accident source term and atmospheric dispersion factors are an input to analyses that evaluate the radiological consequences. Some of the proposed changes do affect the design or manner in which the facility is operated following an accident; however, the proposed changes do not involve a revision to the design or manner in which the facility is operated that could increase in the probability of an accident previously evaluated of a DBA discussed in Chapter 15 of the FSAR.

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

The structures, systems and components affected by the proposed changes act as mitigators to the consequences of accidents. Based on the revised analyses, the proposed changes do revise certain performance requirements; however, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter 15 of the FSAR.

Plant-specific radiological analyses have been performed using the AST methodology and new atmospheric dispersion factors. Based on the results of these analyses, it had been demonstrated that the CRHE dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the offsite doses are well within acceptable limits. This guidance is presented in 10 CFR 50.67, RG [Regulatory Guide] 1.183, and Standard Review Plan [SRP] Section 15.0.1.

Therefore, the proposed amendment does not result in a significant increase in the consequences of any previously evaluated accident.

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

Response: No.

Implementation of AST and the associated proposed TS changes and new atmospheric dispersion factors do not alter or involve any design basis accident initiators. These changes do not affect the design function or mode of operation of structures, systems and components in the facility prior to a postulated accident. Since structures, systems and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change.

Licensing basis changes are proposed and justified to credit use of the SLC [Standby Liquid Control] System to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss of coolant accident. There are new required manual operator actions associated with the SLC System that are not currently considered in the SSES design basis. Operator training will be updated to reflect the new manual operator actions for the pH control function of the of the SLC System as defined in the TS Section 3.1.7. These changes are not significant because the operators are already trained for the operation of the SLC System. Procedural changes are mostly limited to the timing of SLC initiation and termination. In addition, no new hardware changes are necessary to use SLC in this new functional mode.

Licensing basis changes are proposed and justified for the operation of the CREOASS [control room emergency outside air supply system] to respond to NRC Generic Letter 2003–01 and TSTF [Technical Specification Task Force] 448. No new hardware changes are necessary to implement these changes. Since CREOASS will not be operated differently as a result of these changes, no new failure modes are created by these changes.

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

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

Response: No.

The results of the accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, RG 1.183, and SRP 15.0.1. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

Changes to the SLC System to credit use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution and the CREOASS to address NRC Generic Letter 2003–01 and TSTF–448 [to] improve the margin of safety.

New offsite and Control Room atmospheric dispersion factors (χ/Qs) based on site specific meteorological data, calculated in accordance with the guidance of RGs 1.145 and 1.194, utilizes more recent data and improved calculational methodologies.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in 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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: June 7, 2006.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 3/4.6.4, ''Combustible Gas Control,'' TS 3/4.6.4.1, ''Hydrogen Analyzers,'' and TS 3/4.6.4.2, ''Electric Hydrogen Recombiners.'' The changes would be consistent with NRC-approved TS Task Force (TSTF) Standard Technical Specification (STS) Change Traveler number TSTF–447, Revision 1, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors,'' as part of the Consolidated Line Item Improvement Process (CLIIP).

The NRC staff issued a notice of availability of ''Model Application Concerning Technical Specification Improvement To Eliminate Hydrogen Recombiner Requirement, and Relax the Hydrogen and Oxygen Monitor Requirements for Light Water Reactors

Using the Consolidated Line Item Improvement Process (CLIIP)'', in the **Federal Register** on September 25, 2003 (68 FR 55416). The notice included a model safety evaluation (SE), a model no significant hazards consideration (NSHC) determination, and a model application.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee presented an analysis of NSHC by endorsing the model NSHC determination for TSTF–447 which was published in the **Federal Register** on September 25, 2003 (68 FR 55416) as follows:

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 [Regulatory Guide] 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 SAMGs [Severe Accident Management Guidelines], the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

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.

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

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

NRC Acting Branch Chief: Brooke D. Poole.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the

NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendment: January 31, 2006, as supplemented by letter dated July 12, 2006.

Brief description of amendment: The amendment changed the technical specifications to address issues related to an inconsistency that was inadvertently introduced during conversion to improved technical specifications when ''1 per room'' replaced ''2'' as the required channels per trip system for the reactor water cleanup area ventilation differential temperature—high isolation function.

Date of issuance: August 7, 2006.

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

Amendment No.: 173.

Facility Operating License No. NPF–43: Amendment revised the Technical Specifications, Surveillance Requirements, and License.

Date of initial notice in ***Federal Register:*** March 14, 2006 (71 FR 13171). The July 12, 2006, supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2006.

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: September 13, 2005.

Brief description of amendment: The amendment revised the Millstone Power Station, Unit No. 3 Technical Specification (TSs) temperature requirement for the reactivity control system rod drop time test.

Date of issuance: August 15, 2006.

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

Amendment No.: 231.

Facility Operating License No. NPF–49: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** October 25, 2005 (70 FR 61656).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 24, 2005, revised by letter dated May 2, 2006.

Brief description of amendment: The Technical Specification amendment deleted the requirements for NRC approval of the engineering evaluation justifying continued reactor operation with safety relief valve (SRV) discharge pipe temperature exceeding the limit.

Date of issuance: August 4, 2006.

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

Amendment No.: 222.

Facility Operating License No. DPR–35: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** August 16, 2005 (70 FR 48205).

The licensee originally requested for deletion of TS 3.6.D.3 in their submittal dated May 24, 2005. The NRC staff found this unacceptable. Therefore, the licensee revised the original application by letter dated May 2, 2006, reducing the scope of the application as originally noticed. Hence, there is no change to the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 4, 2006.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: July 21, 2005, as supplemented on February 15, May 3, and June 2, 2006.

Brief description of amendment: The amendment modified the Waterford 3 Technical Specification (TS) to revise the existing steam generator (SG) tube surveillance program to be consistent with TS Task Force (TSTF) Change TSTF–449, ''Steam Generator Tube Integrity,'' Revision 4, and the model safety evaluation prepared by the NRC and published in the **Federal Register** notice on March 2, 2005 (70 FR 10298). In this regard, the scope of the application includes changes to the definition of leakage, changes to the primary-to-secondary leakage requirements, changes to the SG tube surveillance program, changes to the SG reporting requirements, and associated changes to the TS Bases.

Date of issuance: July 31, 2006.

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

Amendment No.: 204.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications and the Facility Operating License.

Date of initial notice in ***Federal Register:*** October 25, 2005 (70 FR 61659). The February 15, May 3, and June 2, 2006, supplemental letters contained 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 finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 2006.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: October 25, 2005.

Brief description of amendment: The amendment modified the Waterford 3 Technical Specification 3.1.1.4, ''Minimum Temperature for Criticality,'' to raise the minimum temperature for criticality from the current value of ≥520 °F to ≥533 °F. Changes were also proposed to the associated Action statement to reflect the increase in temperature and to replace the current statement in Surveillance Requirement 4.1.1.4 with wording consistent with NUREG–1432, ''Standard Technical Specifications—Combustion Engineering Plants.''

Date of issuance: July 31, 2006.

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

Amendment No.: 205.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 6, 2005 (70 FR 72672).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 5, 2004, as supplemented by letters dated January 17, October 10, and November 2, 2005 and May 30, 2006.

Brief description of amendment: This amendment revised the technical specifications (TSs) for instrumentation setpoints, allowable values, and calibration requirements based on updated calculations and reviews, and add a definition of ''annual'' frequency for use in the TSs.

Date of issuance: August 9, 2006.

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

Amendment No.: 275.

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

Date of initial notice in ***Federal Register****:* June 8, 2004 (69 FR 32074). The January 17, October 10, and November 2, 2005 and May 30, 2006, supplements contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 28, 2005, as supplemented by letter dated January 12, 2006.

Brief description of amendments: The amendment revised Technical Specifications (Tss) 1.1, ''Definitions,'' 3.4, ''Reactor Coolant System [RCS],'' and 5.7, ''Reporting Requirements,'' to relocate the RCS pressure-temperature curves and limits from the TSs to a licensee-controlled document identified as the Pressure and Temperature Limit Report.

Date of issuance: July 13, 2006.

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

Amendment Nos.: Unit 2–203; Unit 3–195.

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

Date of initial notice in ***Federal Register:*** March 1, 2005 (70 FR 9996). The January 12, 2006, supplemental letter 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 July 13, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 19, 2005, as supplemented by letter dated June 9, 2006.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) Limiting Conditions for Operation 3.3.1, ''Reactor Trip System Instrumentation,'' and TS Surveillance Requirement (SR) 3.2.4.2, ''Quadrant Power Tilt Ratio (QPTR).'' The amendments revise TS 3.3.1, Condition D and the note in SR 3.2.4.2 to clarify when a flux map for QTPR is required.

Date of issuance: August 15, 2006.

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

Amendment Nos.: 143 and 123.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register****:* March 14, 2005 (71 FR 13179). The supplement dated June 9, 2006, provided clarifying information that did not change the scope of the September 19, 2005, application nor the initial proposed no significant hazards consideration determination.

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

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: August 10, 2005.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) to relocate diesel fuel oil testing methods from TS 5.5.13 to a licensee-controlled document, provided clarifications, and corrected the format and typographical errors.

Date of issuance: July 28, 2006.

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

Amendment Nos.: 127 and 127.

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

Date of initial notice in ***Federal Register****:* November 8, 2005 (70 FR 67754).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 28, 2006.

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: September 13, 2005, as supplemented on April 7 and May 23, 2006.

Brief Description of amendments: These amendments revised Technical Specification 5.1, ''Site,'' to redefine the exclusion area boundary as the site boundary.

Date of issuance: August 10, 2006.

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

Amendment Nos.: 249, 248.

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:*** January 3, 2006 (71 FR 156). The April 7 and May 23, 2006, supplements contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 10, 2006.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room (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/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.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. 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, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, 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 identify 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 intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1] Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.
2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.
3. Miscellaneous—does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/requestors shall jointly designate a representative who shall have the authority to act for the petitioners/requestors with respect to that contention. If a petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the authority to act for the petitioners/requestors with respect to that contention.

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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed 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 or 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).

Entergy Nuclear Operations, Inc., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: July 26, 2006.

Description of amendment request: The amendment revised Function 6 [Containment Water Level (Containment Sump)] of Table 3.3.3–1 (''Post Accident Monitoring Instrumentation''), referenced in the Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.3, ''Post Accident Monitoring Instrumentation.'' The revision changed Function 6 to specify 2 required channels for the Containment Sump water level instrumentation instead of 3 channels.

Date of issuance: July 28, 2006.

Effective date: As of its date of issuance, and shall be implemented prior to the expiration of the current 7-day allowed outage time for inoperable containment sump water level channels, which was entered on July 24, 2006.

Amendment No.: 249.

Facility Operating License No. DPR–26: The amendment revised the TS and License.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a Safety Evaluation dated July 28, 2006.

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Richard J. Laufer.

Dated at Rockville, Maryland, this 21st day of August 2006.

For the Nuclear Regulatory Commission.

Cornelius F. Holden,

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

[FR Doc. 06–7137 Filed 8–28–06; 8:45 am]

BILLING CODE 7590–01–P

[1] To the extent that the applications contain attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

POSTAL RATE COMMISSION

Sunshine Act Meetings

NAME OF AGENCY: Postal Rate Commission.

TIME AND DATE: Monday, August 28, 2006, at 3 p.m.

PLACE: Commission conference room, 901 New York Avenue, NW., Suite 200, Washington, DC 20268–0001.

STATUS: Open.

*Matters to be Considered***:** Consideration of fiscal year 2007 budget and election of vice chairman.

FOR FURTHER INFORMATION CONTACT: Stephen L. Sharfman, General Counsel, at 202–789–6820.

Dated: August 24, 2006.

Steven W. Williams,

Secretary.

[FR Doc. 06–7234 Filed 8–24–06; 4:37 pm]

BILLING CODE 7710–FW–M

RAILROAD RETIREMENT BOARD

Proposed Collection; Comment Request

SUMMARY: In accordance with the requirement of Section 3506(c)(2)(A) of the Paperwork Reduction Act of 1995 which provides opportunity for public comment on new or revised data collections, the Railroad Retirement Board (RRB) will publish periodic summaries of proposed data collections.

Comments are invited on: (a) Whether the proposed information collection is