Week of November 19, 2007

Tuesday, November 20, 2007

9:05 a.m.
Affirmation Session (Public Meeting) (Tentative).
a. Pacific Gas and Electric Co. (Diablo Canyon ISFSI), Docket No. 72–26–ISFSI, San Luis Obispo Mothers for Peace's Contentions and Request for a Hearing Regarding Diablo Canyon Environmental Assessment Supplement (Tentative).
b. Dominion Nuclear North Anna, LLC (Early Site Permit for North Anna ESP Site), LBP–07–9 (June 9, 2007) (Tentative).

Week of November 26, 2007—Tentative

Tuesday, November 27, 2007.

9:30 a.m.
Discussion of Security Issues (Closed—Ex. 1 & 3).
1:30 p.m.
Briefing on Equal Employment Opportunity (EEO) Programs (Public Meeting) (Contact: Sandra Talley, 301 415–8059).

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

Week of December 3, 2007—Tentative

Friday, December 7, 2007

10 a.m.
Discussion of Intragovernmental Issues (Closed—Ex. 1 & 9).
2 p.m.
Briefing on Threat Environment Assessment (Closed—Ex. 1).

Week of December 10, 2007—Tentative

Wednesday, December 12, 2007

9:30 a.m.
Discussion of Management Issues (Closed—Ex. 2).

Week of December 17, 2007—Tentative

There are no meetings scheduled for the Week of December 17, 2007.

Week of December 24, 2007—Tentative

There are no meetings scheduled for the Week of December 24, 2007.

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

Additional Information

''Discussion of Management Issues (Closed—Ex. 2)'' previously scheduled on Thursday, December 13, 2007, at 9:30 a.m. has been postponed.

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

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g. braille, large print), please notify the NRC's Disability Program Coordinator, Rohn Brown, at 301–492–2279, TDD: 301–415–2100, or by e-mail at *REB3@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

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

Dated: November 15, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–5772 Filed 11–16–07; 11:31 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 25, 2007, to November 7, 2007. The last biweekly notice was published on November 6, 2007 (72 FR 62685).

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

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

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

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

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

Within 60 days after the date of publication of this notice, person(s) 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 via electronic submission through the NRC E-Filing system 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 person(s) 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 supports 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 hearing or a petition for leave to intervene must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated in August 28, 2007 (72 FR 49139). The E-Filing process requires participants to submit and serve documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *HEARINGDOCKET@NRC.GOV,* or by calling (301) 415–1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system.

The Workplace Forms Viewer™ is free and is available at *http://www.nrc.gov/site-help/e-submittals/install-viewer.html.* Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.*

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The EIE system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the ''Contact Us'' link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html* or by calling the NRC technical help line, which is available between 8:30 a.m. and 4:15 p.m., Eastern Time, Monday through Friday. The help line number is (800) 397–4209 or locally, (301) 415–4737.

Participants who believe that they have a good cause for not submitting documents electronically must file a motion, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted 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; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii). To be timely, filings must be submitted no later than 11:59 p.m. Eastern Time on the due date.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket, which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp,* unless excluded pursuant to an order of the Commission, the Atomic Safety and Licensing Board, or a presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Non-timely 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 amendment 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.*

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

Date of amendments request: October 17, 2007.

Description of amendments request: The proposed amendment would modify the Technical Specifications (TS) to establish more effective and appropriate action, surveillance, and administrative requirements related to the inoperability of snubbers in accordance with Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) change traveler TSTF–372–A, Revision 4. Specifically, the proposed amendment would add Limiting Condition for Operation (LCO) 3.0.8. The NRC staff issued a ''Notice of Opportunity To Comment on Model Safety Evaluation on Technical Specification Improvement To Modify Requirements Regarding the Addition of LCO 3.0.8 on the Inoperability of Snubbers Using the Consolidated Line Item Improvement Process'' in the **Federal Register** on November 24, 2004 (69 FR 68412). The notice included a model safety evaluation (SE) and a model no-significant-hazards-consideration (NSHC) determination. The NRC staff issued a ''Notice of Availability of Model Application Concerning Technical Specification Improvement To Modify Requirements Regarding the Addition of Limiting Condition for Operation 3.0.8 on the Inoperability of Snubbers Using the Consolidated Line Item Improvement Process'' in the **Federal Register** on May 4, 2005 (70 FR 23252). The notice included a model application, including a revised model SE. In its application dated October 17, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed).

Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

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

The NRC staff has reviewed the analysis adopted by the licensee and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves NSHC.

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

NRC Branch Chief: Mark G. Kowal.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of amendment request: September 14, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) requirements related to control room envelope habitability. The proposed changes include revisions to the control room post-accident recirculation system, the instrument operating conditions for isolation functions, and a control room envelope habitability program. The changes are consistent with TS Task Force (TSTF) Change Traveler TSTF–448–A, Revision 3, ''Control Room Habitability,'' except for the differential pressure surveillance requirements. The availability of this TS improvement was published in the **Federal Register** on January 17, 2007 (72 FR 2022).

In addition to the changes related to TSTF–448–A, the proposed amendment would: (1) Align TS with those delineated in NUREG–1431, Revision 3, ''Standard Technical Specifications, Westinghouse Plants,'' to the extent necessary to adopt TSTF–448–A, including the adoption of the necessary portions of TSTF–51–A, Revision 2, ''Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations,'' and TSTF–287–A, Revision 5, ''Ventilation System Envelope Allowed Outage Time,'' (2) add TS for control room radiation monitor R–23 (ventilation system air monitor), and (3) reformat or clarify current TS.

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?

No.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed changes do not prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. This is a revision to the TS for the control room post-accident recirculation system and control room isolation function, which are mitigation systems designed to minimize unfiltered air in-leakage into the control room envelope and to filter the control room envelope atmosphere to protect the control room envelope occupants following accidents previously analyzed. An important part of the system is the control room envelope boundary. The control room envelope post-accident recirculation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not significantly increased.

Establishing operability requirements for SSCs, performing tests and implementing programs that verify the integrity of the control room envelope boundary and control room envelope habitability ensure that the mitigation features are capable of performing their assumed functions. Therefore, the consequences of any accident previously evaluated are not significantly increased.

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

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

No.

The proposed changes will not significantly change the requirements of the control room envelope ventilation system or its function during accident conditions. No new or different accidents result from performing the new surveillance or following the new program. 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 significant change in the methods governing normal plant operation. The proposed changes are consistent with the safety analysis assumptions including the revised gas decay tank and volume control tank rupture analysis and current plant operating practice.

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

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

No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis for an unacceptable period without compensatory measures. The proposed changes do not significantly affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

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

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

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, Riverside 2, Richmond, VA 23219.

NRC Acting Branch Chief: Travis L. Tate.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of amendment request: October 2, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Sections 3.7, ''Auxiliary Electrical Systems'' and 4.6, ''Periodic Testing of Emergency Power System,'' to change the testing requirements for ensuring operability of the remaining operable emergency diesel generator (EDG) when the other EDG is inoperable. In addition, the proposed amendment would add a new specification when two EDGs are inoperable and revise the surveillance requirements for the EDGs.

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?

No.

The proposed amendment would clarify testing requirements for the operable EDG, when one EDG is inoperable, and limit testing to only the intended purpose of the requirement. The intended purpose of the testing requirement is to provide reasonable assurance that when an EDG is inoperable, the opposite EDG is operable. The proposed change does not affect the initiators of analyzed events or the assumed mitigation of accident or transient events. Specifically, testing of the remaining operable diesel will still occur unless evaluation of the inoperable EDG confirms that its failure is not attributable to a common cause failure mechanism. Furthermore, the proposed change clarifies the surveillance testing necessary to give reasonable assurance of operability and restricts the amount of time to perform the testing (i.e. with two inoperable EDGs) to two hours. This ensures no significant increase in the probability of a loss-of-power during the period of the confirming surveillance concurrent with an opposite train inoperable EDG. Elimination of unnecessary testing by acceptable evaluation of the operable EDG reduces component wear and promotes overall EDG reliability and availability. Clarification of required testing and restriction in the amount of time to complete the surveillance to confirm operability, reduces the probability and significance of common mode failures.

The proposed amendment would also add a new specification allowing two EDGs to be inoperable for up to two hours. This change does not significantly increase the initiators of analyzed events or the assumed mitigation of any accidents or transients. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment does not involve a physical alteration of the plant or a change in the methods used to respond to any evaluated plant accident. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. Only a surveillance test clarification and limited two-hour action statement have been added to permit testing of the opposite train, operable EDG. Although the diesel generators will be tested in a different manner, the proposed changes will improve the availability and reliability of the diesel generators without creating the possibility of a new or different kind of accident from any accident previously evaluated. Furthermore, there is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Since the diesel generators will continue to be operated in the same manner and the proposed test protocol will improve diesel generator availability and reliability, no new failure modes are introduced by the proposed amendment.

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

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

Response: No.

The proposed amendment would add a TS allowing two EDGs to be inoperable for up to two hours before the plant must be shut down in a controlled manner. Allowing two EDGs to be inoperable for this limited period of time, while the normal offsite power source remains available, is consistent with Regulatory Guide 1.93 and not considered to be a significant reduction in a margin of safety.

Station operations and EDG surveillance requirements are not adversely affected by the proposed change. Furthermore, the proposed amendment does not adversely impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. The proposed amendment adds provisions to reduce EDG wear and increase availability.

Therefore, the proposed amendment to the KPS [Kewaunee Power Station] TS does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., Counsel for Dominion Energy Kewaunee, Inc., 120 Tredegar Street, Richmond, VA 23219.

NRC Acting Branch Chief: Travis L. Tate.

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

Date of amendment request: October 16, 2007.

Description of amendment request: The proposed amendments would revise the Technical Specifications to accommodate plant modifications that will address water hammer concerns described in Generic Letter 96–06, ''Assurance of Equipment Operability and Containment Integrity During Design-Basis Conditions,'' dated September 30, 1996.

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

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

The requested license amendment seeks approval for the Low Pressure Service Water Reactor Building Waterhammer Prevention System that is being added to the design of the three Oconee Units and the associated revised Technical Specifications. The Low Pressure Service Water Reactor Building Waterhammer Prevention modification will provide a combination passive and automatic means to isolate the Low Pressure Service Water flow stream to the Reactor Building Cooling Units, Reactor Building Auxiliary Coolers, and Reactor Coolant Pump Motor Coolers on a loss of Low Pressure Service Water flow that can lead to a waterhammer should the Low Pressure Service Water system become depressurized.

New check valves and air operated valves are added into an Engineered Safeguards flowpath. The existing Low Pressure Service Water header that discharges from the Reactor Building Cooling Units is to be modified by separating it into two headers and then joining back into a common header. Each header will contain two air operated valves. The Waterhammer Prevention System maintains the Low Pressure Service Water System inside containment water solid during a Loss of Offsite Power such that voids, which could later collapse, cannot form. The Waterhammer Prevention System will eliminate an Operable but degraded/non-conforming condition associated with waterhammers.

The design of the proposed modification and its associated Technical Specifications will provide means to assure that the Low Pressure Service Water Reactor Building Waterhammer Prevention System operates at a performance level necessary to provide for safe operation of the Low Pressure Service Water system following installation on each of the three Units. The system is designed such that a single active failure will not prevent the system from preventing a waterhammer event if power is lost to the Low Pressure Service Water pumps (e.g., Loss of Offsite Power), nor will a single active failure prevent the Engineered Safeguards flowpath from being available if needed during a Loss of Coolant Accident or Main Steam Line Break. Evaluations have been performed to assure that the risk of adding new hardware is acceptable.

Therefore, the addition of this modification and associated Technical Specifications does not significantly increase the probability or consequences of any accident previously evaluated.

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

The proposed Low Pressure Service Water Reactor Building Waterhammer Prevention Modification and its associated Technical Specifications will provide a means to assure the mechanical and electrical components operate at a performance level necessary to provide for safe operation of the modified Low Pressure Service Water system flow to the Reactor Building Cooling Units, Reactor Building Auxiliary Coolers and Reactor Coolant Pump Motor Coolers.

The change enhances the plant design by eliminating the possibility of significant waterhammers that occur on a loss of Low Pressure Service Water flow to the above components.

The modification does not add any new single active failures that would prevent the Low Pressure Service Water System from supplying cooling water to the Reactor Building Cooling Units. The Reactor Building Cooling Units will be isolated briefly during an Engineered Safeguards event; however, the flow path will be restored before cooling is required following the event. Since cooling was previously not available until after power restoration following a Loss of Offsite Power, there is no change in system response regarding Low Pressure Service Water flow through the Reactor Building Cooling Units when compared to the previous design.

Therefore, the proposed modification and associated Technical Specifications will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

The proposed change does not adversely affect any plant safety limits, setpoints, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or Containment Operability. The Reactor Building Cooling Units will be isolated briefly during an Engineered Safeguards event; however, the flow path will be restored before cooling is required following the event.

Since cooling is currently not available until after power restoration following a Loss of Offsite Power, there is no change in system response regarding Low Pressure Service Water flow through the Reactor Building Cooling Units when compared to the previous design.

The modification mitigates significant waterhammers in the Low Pressure Service Water piping to the Reactor Building Cooling Units and Reactor Cooling Pump Motor Coolers. The change will maintain the ability to provide Low Pressure Service Water flow to safety related loads following Loss of Offsite Power events.

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: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: October 22, 2007.

Description of amendment request: The proposed amendments would revise the Technical Specifications to accommodate the use of AREVA NP Mark–B–HTP fuel.

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

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

The proposed revisions to the technical specifications and to Duke's NRC-approved methodology reports support the use of the AREVA NP Mark–B–HTP fuel design. The methodology will be approved by the NRC prior to plant operation with the new fuel. The proposed safety limit ensures that fuel integrity will be maintained during normal operations and anticipated operational transients. The core operating limits report will be developed in accordance with the approved methodology. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

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

The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The BHTP correlation is not an accident/event initiator. No new initiating events or transients result from the use of the BHTP correlation or the related safety limit change.

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

The proposed safety limit value has been established in accordance with the methodology for the BHTP correlation to ensure that the applicable margin of safety is maintained (i.e. there is at least 95% probability at a 95% confidence level that the hot fuel rod does not experience DNB). The other reactor core safety limits will continue to be met by analyzing the reload using NRC approved methods and incorporation of resultant operating limits into the Core Operating Limits Report (COLR).

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

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: August 30, 2007.

Description of amendment request: The proposed amendment would modify Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2) Technical Specification (TS) requirements related to control room envelope habitability in TS 3.7.10, ''Control Room Emergency Ventilation System (CREVS)'' and TS Section 5.5, ''Administrative Controls—Programs and Manuals.'' This change is consistent with Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Change Traveler TSTF–448, Revision 3. The availability of this TS revision was announced in the **Federal Register** on January 17, 2007 (72 FR 2022) 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 adopted by the licensee 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 does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Mark G. Kowal.

FPL Energy, Point Beach, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: October 1, 2007.

Description of amendment request: The proposed amendments would revise the accident source term in the design-basis radiological consequences analyses and the associated Technical Specifications (TSs), pursuant to Section 50.67 of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50.67). The proposed amendments would revise the licensing basis of Point Beach Nuclear Plant, Units 1 and 2 (PBNP) to support a full-scope application of an Alternative Source Term (AST) methodology. The AST methodology will modify PBNP's licensing bases by: (1) Replacing the current accident source term with an AST as described in 10 CFR 50.67 for design-basis accidents (DBA) radiological consequences, and (2) establishing the 10 CFR 50.67 Total Effective Dose Equivalent (TEDE) dose limits as acceptance criteria for the radiological consequences of DBAs.

TS changes associated with the AST methodology change are: TS 1.1, a reduction in the definition of the maximum allowable containment leak rate. TS 3.4.16, the specific activity of the reactor coolant is revised for dose equivalent iodine. TS 3.7.9, a new mode of operation for the Control Room Emergency Filtration System (CREFS), which will allow operation of the CREFS with filtered outside and filtered recirculated air.

TS 3.7.13, the specific activity of the secondary coolant is revised for dose equivalent iodine. In addition, a modification to the residual heat removal system, containment spray and their support systems, will be made to support operation of the containment spray system during containment spray recirculation.

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

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

Response: No.

The results of the applicable radiological design basis accident (DBA) re-evaluation demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory limits and guidance provided by the NRC in 10 CFR 50.67 and Regulatory Guide 1.183 for alternative source term (AST) methodology. The AST is an input to calculations used to evaluate the consequences of an accident and does not by itself affect the plant response or the actual pathway of the activity released from the fuel. It does, however, better represent the physical characteristics of the release such that appropriate mitigation techniques may be applied.

The change from the original source term to the new proposed AST is a change in the analysis method and assumptions and has no effect on accident initiators or causal factors that contribute to the probability of occurrence of previously analyzed accidents. Use of an AST to analyze the dose effect of DBAs shows that regulatory acceptance criteria for the new methodology continues to be met. Changing the analysis methodology does not change the sequence or progression of the accident scenario.

The proposed Technical Specification changes reflect the plant configuration that will support implementation of the AST analyses. The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. The operation of various filtration systems, the residual heat removal and the containment spray system, including associated support systems, has been considered in the evaluations for these proposed changes. While the operation of these systems does change with the implementation of an AST, the affected systems are not accident initiators, and application of the AST methodology itself, is not an initiator of a design basis accident.

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

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

Response: No.

As described in Item 1 above, the changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will support implementation of the new methodology. No new or different accidents result from utilizing the proposed changes. Although the proposed changes require modifications to the control room emergency ventilation system, as well as modifications to the residual heat removal system and containment spray system, these changes will not initiate a new or different kind of accident since they are related to system capabilities that provide protection from accidents that have already occurred. As a result, no new failure modes are being introduced that could lead to different accidents. These changes do not alter the nature of events postulated in the Updated Final Safety Analysis Report nor do they introduce any unique precursor mechanisms.

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

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

Response: No.

As described in Item 1, the changes proposed in this license amendment involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will support implementation of the new methodology. Safety margins and analytical conservatisms have been evaluated and have been found to be acceptable. The analyzed events have been carefully selected and, with plant modifications, margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The proposed changes continue to ensure that the dose consequences of DBAs at the exclusion area and low population zone boundaries and in the control room are within the corresponding acceptance criteria presented in RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological consequences of these accidents is provided by meeting the applicable regulatory limits, which are set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

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

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

Attorney for licensee: Mr. Antonio Fernandez, Senior Attorney, FPL Energy Point Beach, LLC P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Acting Branch Chief: Travis L. Tate.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Oswego County, New York

Date of amendment request: September 27, 2007.

Description of amendment request: The proposed amendment would revise the operability requirements contained in Technical Specification (TS) Section 3.2.7, ''Reactor Coolant System Isolation Valves,'' and associated requirements contained in TS Section 3.6.2, ''Protective Instrumentation.'' The proposed changes would modify the conditions for which reactor coolant system isolation valves (RCSIVs) and associated isolation instrumentation must be operable to include the hot shutdown reactor operating condition (i.e., when fuel is in the reactor vessel and the reactor coolant temperature is greater than 212 °F). In addition, new requirements are proposed to require that the RCSIVs in the shutdown cooling (SDC) system and associated isolation instrumentation be operable during the cold shutdown reactor operating condition (fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212 °F) and the refueling reactor operating condition (i.e., when fuel is in the reactor vessel and the reactor coolant temperature is less than 212 °F). These proposed changes will require operability of RCSIVs during conditions other than the power operating condition, and are similar in concept to primary containment isolation valve operability requirements contained in NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4.'' Lastly, TS Section 3.6.2 (Table 3.6.2b) would be revised to delete unnecessary operability requirements for the cleanup system and SDC system high area temperature isolation instrumentation, consistent with the proposed revisions to the RCSIV operability requirements.

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

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

Response: No.

The proposed changes provide more stringent requirements for operation of NMP1. These include requiring operability of RCSIVs and associated isolation instrumentation during the hot shutdown condition and requiring RCSIVs in the SDC system and associated instrumentation to be operable during the cold shutdown and refueling operating conditions. Requiring RCSIV operability during the hot shutdown operating condition ensures that reactor coolant loss in the event of a rupture of a line connected to the reactor coolant system (RCS) is minimized, and the release of radioactive material to the environment is consistent with the assumptions used in the analyses for design basis accidents. Requiring operability of the RCSIVs in the SDC system during the cold shutdown and refueling operating conditions provides protection against potential draining of the reactor vessel through the SDC system during shutdown conditions, which is when the SDC system is normally operated.

In addition, operability requirements for the cleanup system and SDC system high area temperature isolation instrumentation are revised to be consistent with the proposed revisions to the RCSIV operability requirements and with NUREG–1433. The high area temperature isolation instrumentation need not be operable in the cold shutdown and refueling conditions, since the probability and consequences of design basis accidents are reduced due to the pressure and temperature limitations of these operating conditions. Also, system isolation on high area temperature would likely not occur in the event of system leakage or line break since RCS temperature during the cold shutdown and refueling conditions is typically maintained below the high area temperature isolation setpoints (190°F for the cleanup system area and 170°F for the SDC system area).

The revised operability requirements for the RCSIVs and associated isolation instrumentation do not result in operation that would make an accident more likely to occur and do not alter assumptions relative to mitigation of a previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes to the TS operability requirements for the RCSIVs and associated isolation instrumentation do not alter or involve any design basis accident initiators. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes do impose different RCSIV operability requirements that are more stringent than existing requirements, and incorporate RCSIV isolation instrumentation operability requirements that are consistent with the RCSIV requirements and with NUREG–1433. These changes continue to be consistent with the assumptions in the safety analyses and licensing basis. Therefore, the proposed changes 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 proposed changes to the TS operability requirements for the RCSIVs and associated isolation instrumentation ensure that RCSIV closure will occur when required to mitigate the consequences of design basis accidents. The proposed changes also ensure that SDC system isolation can be accomplished to protect against potential draining of the reactor vessel through the SDC system during shutdown conditions, which is when the SDC system is normally operated. The imposition of these revised RCSIV operability requirements either has no impact on or increases the margin of plant safety. The plant responses to accidents will not be adversely affected, and the accident mitigation equipment will continue to function as assumed in the accident analyses. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: September 19, 2007.

Description of amendment request: The proposed amendment would revise NMP2 Limiting Condition for Operation (LCO) 3.10.1 to expand its scope to include provisions for temperature excursions greater than 200 °F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in Mode 4. This change is consistent with Nuclear Regulatory Commission (NRC)-approved Revision 0 to Technical Specification (TS) Task Force (TSTF) Change Traveler, TSTF–484, ''Use of TS 3.10.1 for Scram Time Testing Activities.'' The availability of this TS revision was announced in the **Federal Register** on October 27, 2006 (71 FR 63050) as part of the consolidated line item improvement process. The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

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

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

Technical Specifications currently allow for operation at greater than [200]°F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact the probability or consequences of an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Technical Specifications currently allow for operation at greater than [200]°F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. No new operational conditions beyond those currently allowed by LCO 3.10.1 are introduced. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Technical Specifications currently allow for operation at greater than [200]°F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact any margin of safety. Allowing completion of inspections and testing and supporting completion of scram time testing initiated in conjunction with an inservice leak or hydrostatic test prior to power operation results in enhanced safe operations by eliminating unnecessary maneuvers to control reactor temperature and pressure. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: September 25, 2007.

Description of amendment request: The proposed amendment would revise the MNGP licensing basis to incorporate the results of a revised small-break loss-of-coolant accident (LOCA) analysis to determining the Low Pressure Coolant Injection (LPCI) loop select logic minimum detectable break area. This analysis showed that a small break, rather than the current large recirculation line break LOCA, would become the limiting accident with respect to peak cladding temperature (PCT). In conjunction with this proposed new licensing basis analysis, the licensee proposed to revise the Table 3.3.5.1–1 (regarding emergency core cooling system instrumentation) of the Technical Specifications (TS) as follows: (1) change the allowable value from the current 24 inch water column to 100 inch water column for Function 2.j, ''Recirculation Riser Differential Pressure—High (Break Detection);'' and (2) change the associated channel calibration frequency Surveillance Requirement (SR) from a nominal 12-month to a 24-month interval.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC). The NRC staff reviewed the licensee's analysis, and has performed its own as follows:

1. Do the proposed changes involve a significant increase in the probability or

consequences of an accident previously evaluated?

No. The proposed changes to the PCT licensing basis and the TS do not involve a physical alteration of the plant, i.e., no design change to plant system, and no new or different type of equipment will be installed. The proposed PCT change is an analysis result which is within regulatory acceptance limits, and the proposed TS changes reflect the revised analysis. Thus, the proposed changes affect only parameters assumed for certain analyses, but do not adversely affect accident initiators, precursors, plant design, configuration, or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation capability, or radiological consequences of any accident previously evaluated. Furthermore, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, and do not significantly increase individual or cumulative occupational/public radiation exposures. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any 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 changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal plant operation. The requirements in the TS will continue to assure operation of the plant within its design specifications and safety limits. Therefore, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed amendment would only change the analysis of record LOCA PCT, the allowed value of an instrument function, and its associated SR frequency. There will be no modification of any TS limiting condition for operation, no change to any limit on previously analyzed accidents, no change to how previously analyzed accidents or transients would be mitigated, no change in any methodology used to evaluate consequences of accidents, and no change in any operating procedure or process. Therefore, the proposed amendment does not entail a significant reduction in a margin of safety.

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

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

NRC Acting Branch Chief: Clifford G. Munson.

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

Date of amendment request: October 5, 2007.

Description of amendment request: The proposed amendment requests a change to Technical Specification 3.7(1)ci, ''Emergency Power Periodic Test,'' related to the surveillance testing of the Fort Calhoun Station emergency diesel generators (DGs) to support a modification to the DG start circuitry.

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

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

Response: No.

The removal of the anticipatory (idle speed) diesel generator (DG) start signal on a reactor protective system (RPS) reactor trip does not adversely affect the design function of the DGs and thus is not an initiator of any previously evaluated accidents.

No Updated Safety Analysis Report (USAR) accident analyses take credit for the anticipatory (idle speed) DG start following a design basis accident (DBA). The DGs provide emergency power to their respective 4.16 KV [Kilovolt] buses and will continue to do so after the proposed modification is installed. Upon the occurrence of an undervoltage condition on the bus or an engineered safety features (ESF) signal, the modification provides a full speed DG start to achieve rated voltage and frequency. The safety function of the DGs is not altered by the installation of the modification. The associated Technical Specification (TS) change allows surveillance testing to reflect the way that the DGs start and load onto their respective buses following the modification.

Deletion of a footnote containing historical information pertaining to a one-time surveillance interval extension and the punctuation correction are administrative changes. These administrative changes do not increase the probability or consequences of any accident previously evaluated.

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

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

Response: No.

The removal of the anticipatory (idle speed) diesel generator (DG) start signal on an RPS reactor trip does not adversely affect the design function of the DGs and thus does not create the possibility of a new or different kind of accident. There are no USAR accident analyses which take credit for the anticipatory (idle speed) DG start following a DBA. The DGs provide emergency power to their respective 4.16 KV buses and will continue to do so after the proposed modification is installed. Upon the occurrence of an undervoltage condition on the bus or an ESF signal, the modification provides a full speed DG start to achieve rated voltage and frequency. The safety function of the DGs is not altered by the installation of this modification. The associated TS change allows surveillance testing to reflect the way that the DGs start and load onto their respective buses following the modification.

Deletion of a footnote containing historical information pertaining to a one-time surveillance interval extension and the punctuation correction are administrative changes that do not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

Response: No.

The removal of the anticipatory (idle speed) diesel generator (DG) start signal on an RPS reactor trip does not adversely affect the design function of the DGs and thus does not involve a significant reduction in a margin of safety. There are no USAR accident analyses which take credit for the anticipatory (idle speed) DG start following a DBA. The DGs provide emergency power to their respective 4.16 KV buses and will continue to do so after installation of the proposed modification. Upon the occurrence of an undervoltage condition on the bus or an ESF signal, the modification provides a full speed DG start to achieve rated voltage and frequency. The safety function of the DGs is not altered by the installation of this modification. The associated TS change allows surveillance testing to reflect the way that the DGs will start and load onto their respective buses following the modification.

Deletion of a footnote containing historical information pertaining to a one-time surveillance interval extension and the punctuation correction are administrative changes that do not reduce a margin of safety.

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

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: October 12, 2007.

Description of amendment request: The proposed amendment would modify the Fort Calhoun Station, Unit 1 design and licensing basis to increase the shutdown cooling (SDC) system entry temperature from 300 °F to 350 °F (cold leg), and the SDC entry pressure from 250 psia to 300 psia (indicated at the pressurizer). Additionally, the licensee proposes to change to the Updated Safety Analysis Report (USAR) described design methodology applied to the SDC heat exchangers.

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

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

Response: No.

The shutdown cooling (SDC) system provides flow to the reactor during long term cooling mode following a large break loss-of-coolant accident (LOCA). In addition, the SDC system can supply cooled sump water to the high pressure safety injection (HPSI) pumps for long term core cooling. The SDC system is also designed to reduce the temperature of the reactor coolant system (RCS) from 300 °F to refueling temperature within 24 hours and to maintain the proper RCS temperature during refueling. As such, the SDC system is not an initiator for any accident previously evaluated.

The proposed change to increase the SDC entry temperature from 300 °F to 350 °F affects the inputs to the analysis of the Boron Dilution Incident.

However, re-analysis of this accident with the increased temperature does not result in an increase in the probability of the accident. The proposed increase in SDC system design and operating temperature and pressure has been evaluated for affects on system piping and components using appropriate codes and standards. The proposed changes do not introduce any failure mechanisms that would initiate a previously analyzed accident. Therefore, the proposed change to uprate the SDC system entry conditions does not result in a significant increase in the probability of a previously evaluated accident.

The potential effect of the proposed change on the consequences of a previously evaluated accident has been considered. Re-analysis of the Boron Dilution Incident with the proposed increased SDC entry temperature does not result in an increase in the consequences of the accident.

In addition, although an increase in the SDC system leakage test pressure is proposed, the leakage test acceptance criteria (i.e., maximum permitted leakage per hour) will not be affected. Therefore, the limit on post-accident leakage to atmosphere from the SDC system is unchanged. The proposed increase in SDC system design and operating temperature and pressure does not affect the redundancy or availability of the SDC system. The design functions of the system are not affected by the proposed change. Therefore, the SDC system will still be capable of performing the safety functions needed to mitigate the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed change alters the SDC system entry conditions and increases the system leakage test pressure. In the current design, the SDC system has been excluded from consideration as a pipe rupture initiator since it is not normally in operation. It is used for plant shutdown and startup, and for accident mitigation. With the proposed change, the operating modes of the system will not be affected. The proposed change increases the RCS temperature and pressure at which the SDC system can be placed in service during shutdown (or removed from service during startup), but the RCS, SDC, and other plant systems are not operated in a different manner. The increased heat load on the component cooling water (CCW) system resulting from normal operation of the SDC at increased SDC temperatures has been evaluated. The increased normal operating heat load has been determined to be bounded by the post-accident CCW heat load. Any adjustments to the cooldown rate needed to accommodate the increased SDC entry temperature will be performed using approved procedures consistent with current practice and would not require operating the plant in a different manner.

The RCS cooldown rate limitations in the Technical Specifications (TS) are not affected by the proposed change. In addition, adjustments of CCW heat loads to maintain required CCW inlet temperatures for the SDC (Low Pressure Safety Injection (LPSI)) pump coolers, when operating at the increased SDC entry temperature, will be in accordance with plant procedures and within existing system capabilities. The low temperature overpressurization (LTOP) analysis has been revised for the proposed change. However, there are no effects on existing LTOP setpoints or operating limitations, other than the proposed change to TS 2.1.1(11)(b), which states that the unit cannot be placed on shutdown cooling until the RCS has been cooled to ≤ 350 °F. The proposed change in SDC operating limitations does not introduce the possibility of new or different equipment malfunctions or accident precursors.

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

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

Response: No.

The margins of safety are established through design parameters, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change increases the SDC system entry conditions for plant shutdown, startup and following postulated accidents, and the SDC system leakage test pressure. However, the accident mitigation function and post-accident operation of the system is not affected. The operating limits on temperature and pressure will remain below the design temperature and pressure for the system. The time interval for operator action after a postulated boron dilution event with the SDC system in operation is reduced, however, the available time remains greater than the minimum required time interval of 15 minutes. The proposed change does not affect any design or operating parameter or setpoint used in the accident analyses to establish the margin of safety.

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: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: October 15, 2007.

Description of amendment requests: The proposed amendments would relocate all periodic surveillance frequencies from the technical specifications (TS) and place the frequencies under licensee control in accordance with a new program, the Surveillance Frequency Control Program.

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 involves the relocation of various surveillance test intervals from TSs to a licensee-controlled program and is administrative in nature. The proposed change does not involve the modification of any plant equipment or affect basic plant operation. The proposed change will have no impact on any safety related structures, systems or components. Surveillance test intervals are not assumed to be an initiator of any analyzed event, nor are they assumed in the mitigation of consequences of accidents. The [Surveillance Requirements] themselves will be maintained in the TS along with the applicable Limiting Conditions for Operation (LCOs) and Action statements. The surveillances performed at the intervals specified in the licensee-controlled program will assure that the affected system or component function is maintained, that the facility operation is within the Safety Limits, and that the LCOs are met.

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

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

Response: No.

The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related structure, system, or component performs its function or is tested. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

Therefore, the proposed change 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 proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by relocation of the surveillance test intervals to a licensee-controlled program.

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

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

Attorney for licensee: Jennifer Post, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendments request: October 17, 2007.

Description of amendments request: The proposed amendment would modify the Technical Specifications (TS) to establish more effective and appropriate action, surveillance, and administrative requirements related to the inoperability of snubbers in accordance with Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) change traveler TSTF–372–A, Revision 4. Specifically, the proposed amendment would add Limiting Condition for Operation (LCO) 3.0.8. The NRC staff issued a ''Notice of Opportunity To Comment on Model Safety Evaluation on Technical Specification Improvement To Modify Requirements Regarding the Addition of LCO 3.0.8 on the Inoperability of Snubbers Using the Consolidated Line Item Improvement Process'' in the **Federal Register** on November 24, 2004 (69 FR 68412). The notice included a model safety evaluation (SE) and a model no-significant-hazards-consideration (NSHC) determination. The NRC staff issued a ''Notice of Availability of Model Application Concerning Technical Specification Improvement To Modify Requirements Regarding the Addition of Limiting Condition for Operation 3.0.8 on the Inoperability of Snubbers Using the Consolidated Line Item Improvement Process'' in the **Federal Register** on May 4, 2005 (70 FR 23252). The notice included a model application, including a revised model SE. In its application dated October 17, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

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

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

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

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

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

The NRC staff has reviewed the analysis adopted by the licensee and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves NSHC.

Attorney for licensee: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Branch Chief: Mark G. Kowal.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: October 18, 2007.

Description of amendment request: The proposed amendments to Technical Specification Administrative Controls Section 5.3.1 would revise the training and qualifying education and experience eligibility requirements for certain unit staff positions to correspond to a defined training program. The training program is based on National Academy for Nuclear Training guidance documents (ACADs) as described in the licensee's October 18, 2007, application. The proposed changes will also replace a specific position title with a generic position title for the senior individual in charge of Health Physics. An application that addressed similar issues was previously submitted on October 30, 2006, and notice of that application was provided in the **Federal Register** on July 17, 2007 (72 FR 39084). Due to certain changes in the specifics of the October 18, 2007, application, from those proposed in the October 30, 2006, application, the application is being renoticed in its entirety. This notice supersedes the notice published in the **Federal Register** on July 17, 2007.

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 to Technical Specifications Administrative Controls Section 5.3.1 involves the use of a more generic designation for the unit staff position responsible for Health Physics without reducing the level of authority required for that position. The proposed change also allows the flexibility to use an accredited program for qualifying personnel to fill certain unit staff positions as stipulated in Enclosure 1 [of October 18, 2007, application], which represents an acceptable alternative to the qualification requirements for these positions as currently specified in the Technical Specifications. Since the proposed changes are administrative in nature, they do not involve any physical changes to any structures, systems, or components, nor will their performance requirements be altered. The proposed changes also do not affect the operation, maintenance, or testing of the plant. Therefore, the response of the plant to previously analyzed accidents will not be affected. Consequently, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes to the Technical Specifications will have no adverse impact on the overall qualification of the unit staff. The use of a more generic designation for the unit staff position responsible for Health Physics and the proposed addition [of] a statement to Section 5.3.1 that will reference this letter and the accreditation information for the positions stipulated in Enclosure 1 will allow the use of an accredited program that has been endorsed by the NRC and will ensure the educational requirements and power plant experience for each unit staff position are properly satisfied and will continue to fulfill applicable regulatory requirements. Also, since no change is being made to the design, operation, maintenance, or testing of the plant, no new methods of operation or failure modes are introduced by the proposed changes. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

Response: No

The proposed changes to the Technical Specifications will have no adverse impact on the onsite organizational features necessary to assure safe operation of the plant. Lines of authority for plant operation are unaffected by the proposed changes. Also, the adoption of the more generic designation of the individual responsible for Health Physics will reduce the regulatory burden of having to devote limited resources to process a license amendment whenever a title change for this position is implemented. Accordingly, this reduction in regulatory burden and the proposed addition of a statement to Section 5.3.1 that will reference this letter and the use of accreditation information provided in Enclosure 1, will allow the use of an accredited program endorsed by NRC to qualify certain unit staff positions and will improve organizational flexibility without compromising plant safety. Therefore, the proposed changes do not involve a significant decrease 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: August 28, 2007, as supplemented on October 9, 2007.

Description of amendment request: The proposed amendments would revise the ''Maximum Power Level'' in paragraph 2.C(1) of the Vogtle Electric Generating Plant Facility Operating Licenses NPF–68 and NPF–81 for Unit 1 and Unit 2, respectively. In addition, the amendments would revise the definition of ''Rated Thermal Power (RTP)'' in Technical Specification 1.1 for both units to reflect the change to the Maximum Power Level. The proposed change increases the RTP from 3565 MWt to 3625.6 MWt, resulting in an increase of 1.7% from the current reactor output. This increase in reactor core power level is referred to as a Measurement Uncertainty Recapture (MUR) power uprate.

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?

Operating License—Maximum Power Level and Technical Specification 1.1—Definition of Rated Thermal Power

The increase in Maximum Power Level and Rated Thermal Power (RTP) does not involve a significant increase in the probability or consequences of an accident previously evaluated, because operation at the higher power level will not cause any design or analysis acceptance criteria to be exceeded. As a result, structural and functional integrity of the plant systems is maintained. Power level is an input assumption to the equipment design and accident analyses, but it is not itself an initiator for any transient. Therefore, the probability of occurrence of an accident previously evaluated is not affected.

The radiological consequences of operation at the Measurement Uncertainty Recapture (MUR) power uprate conditions have been assessed. It was concluded that offsite dose predictions remain within the acceptance criteria for each of the accidents affected. Therefore, the consequences of an accident previously evaluated are not increased.

Technical Specification 1.1—Definition of Dose Equivalent Iodine

The proposed change to the definition of dose equivalent iodine (DEI) impacts the reactor coolant activity surveillance and calculations of accident consequences and makes these activities consistent with each other. Neither of these functions affects the probability of any accident previously evaluated.

In order to support the MUR power uprate, the accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) were re-analyzed. As part of this reanalysis, the dose conversion factors (DCFs) were reviewed, and a consistent set of DCFs was used for all re-analyses based on Federal Guidance Report No. 11, as suggested by RIS 2001–19. The results of these re-analyses continue to meet the acceptance limits as currently described in the UFSAR.

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

Technical Specification 3.3.1, Table 3.3.1–1, Function 16—P–9 Setpoint

The revised Power Range Neutron Flux P–9 permissive nominal setpoint and allowable value do not involve a significant increase in the probability or consequences of an accident previously evaluated, because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of any plant system is unaffected. The P–9 permissive function is part of the transient mitigation response and is not itself an initiator for any transient. Therefore, the probability of occurrence of an accident previously evaluated is not affected.

The changes to the P–9 nominal setpoint and allowable value do not affect the integrity of the fission product barriers utilized for the mitigation of radiological dose consequences as a result of an accident. The change continues to ensure that the pressurizer power operated relief valves (PORVs) are not challenged following a turbine trip without a reactor trip which, in turn, minimizes the potential for a release. There are no offsite dose predictions for this transient. Since it has been determined that the transient results are unaffected by the change to the P–9 nominal setpoint and allowable value, it is concluded that the consequences of an accident previously evaluated are not increased.

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

Operating License—Maximum Power Level and Technical Specification 1.1—Definition of Rated Thermal Power

The increase in Maximum Power Level and RTP does not create the possibility of a new or different kind of accident from any previously evaluated, because no new operating configuration is being imposed that will create a new failure scenario, and no new failure modes are being created for any plant equipment. System and component design bases have been reviewed. The proposed change does not have an adverse effect on safety-related systems or components and does not challenge the integrity of any safety-related system. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to determine safe plant operation.

Technical Specification 1.1—Definition of Dose Equivalent Iodine

The proposed change to the definition of Dose Equivalent Iodine (DEI) ensures the reactor coolant activity surveillances are consistent with the assumptions for initial conditions used in the accident analyses. The proposed change does not involve the addition or modification of any plant equipment. Neither does it alter the design, configuration or method of operation of the plant.

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

Technical Specification 3.3.1, Table 3.3.1–1, Function 16—P–9 Setpoint

The revised Power Range Neutron Flux P–9 permissive nominal setpoint and allowable value do not create the possibility of a new or different kind of accident from any previously evaluated, because these changes do not affect accident initiation sequences. No new operating configuration is being imposed by the P–9 nominal setpoint and allowable value changes that will create a new failure scenario. In addition, no new failure modes are being created for any plant equipment. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to determine safe plant operation.

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

Operating License—Maximum Power Level and Technical Specification 1.1—Definition of Rated Thermal Power

The increase in Maximum Power Level and RTP does not involve a significant reduction in a margin of safety, because power level is one of the inherent assumptions that determine the safe operating range defined by the accident analyses, which are in turn protected by the Technical Specifications. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the Technical Specifications. The engineering reviews performed for the MUR power uprate confirmed that the accident analyses criteria are met at the revised value of MPL and RTP. Therefore, the adequacy of the revised Facility Operating Licenses and Technical Specifications to maintain the plant in a safe operating range is also confirmed, and the increase in MPL and RTP do not involve a significant decrease in a margin of safety.

Technical Specification 1.1—Definition of Dose Equivalent Iodine

The proposed change to the definition of dose equivalent iodine (DEI) has the potential to affect the dose consequences offsite and in the control room. However, the results of the re-analyses of the accidents previously evaluated demonstrate the dose consequences at all locations remain within the regulatory acceptance limits, and the margin of safety as defined by 10 CFR 100 and GDC 19 has not been significantly reduced.

Technical Specification 3.3.1, Table 3.3.1–1, Function 16—P–9 Setpoint

The change to the P–9 nominal setpoint and allowable value does not involve a significant reduction in a margin of safety because the margin of safety associated with the P–9 setpoint, as verified by the results of the applicable transient analyses, is within acceptable limits. The adequacy of the revised Technical Specification values to maintain the plant in a safe operating range has been confirmed. Therefore, the change to the P–9 nominal setpoint and allowable value does not involve a significant decrease 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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: August 27, 2007.

Description of amendment request: The amendments would revise the licensee's fire protection program requirements as documented in the licensee's Fire Hazard's Analysis Report. Specifically, the licensee requests the use of reactor operator manual actions in lieu of meeting protection requirements of circuit separation.

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 amendment[s] involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The design function of structures, systems and component[s] are not Impacted by the proposed change. The proposed change involves operator manual actions in response to a fire and will not initiate an event. The proposed actions do not increase the probability of occurrence of a fire or any other accident previously evaluated.

The proposed actions are feasible and reliable and demonstrate that the unit can be safely shutdown in the event of a fire. No significant consequences result from the performance of the proposed actions.

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

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

Response: No.

The design function of structures, systems and component[s] are not impacted by the proposed amendment[s]. The proposed change involves operator manual actions in response to a fire. [It does not] involve new failure mechanisms or malfunctions that can initiate a new accident.

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

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

Response: No.

Adequate time is available to perform the proposed operator manual actions to account for uncertainties in estimates of the time available and in estimates of how long it takes to diagnose and execute the actions. The actions are straightforward and do not create any significant concerns. The actions have been verified that they can be performed through demonstration and they are proceduralized. The proposed actions are feasible and reliable and demonstrate that the unit can be safely shutdown in the event of a fire.

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

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: Thomas G. Hiltz.

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

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

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

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

Date of amendment request: October 22, 2007.

Brief description of amendment request: The proposed amendment would allow an alternate methodology from that previously approved in Topical Report DOM–NAF–3–0.0–P–A, *GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment*, as discussed in the Surry Power Station, Unit Nos. 1 and 2, Updated Final Safety Analysis Report.

*Date of publication of individual notice in **Federal Register***: October 30, 2007 (72 FR 61406).

Expiration date of individual notice: Public comment period expiration date, November 13, 2007; Hearing period expiration date, January 31, 2008.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by email to *pdr@nrc.gov*.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of application for amendment: December 15, 2006.

Brief description of amendment: The amendment incorporates changes to the technical specifications (TSs) associated with previously-approved industry initiatives. The first change relocates the actions for a safety limit violation from the administrative controls TS section to the safety limit TS section and deletes notification requirements, as approved by TS Task Force (TSTF) Change Traveler TSTF–05–A, ''Deletion of Safety Limit Violation Notification Requirements.'' The second change incorporates generic position titles, as approved by TSTF–65–A, ''Use of Generic Titles for Utility Positions,'' and incorporates items approved by Nuclear Regulatory Commission Administrative Letter 95–06, ''Relocation of Technical

Specification Administrative Controls Related to Quality Assurance.''

Date of issuance: October 31, 2007.

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

Amendment No.: 193.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** March 13, 2007 (72 FR 11386) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

Duke Power Company LLC, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendments: November 22, 2006.

Brief description of amendments: The amendment revises the Catawba Unit 1 Facility Operating License (FOL) to add a license condition requiring a specific date by which the modifications to the Emergency Core Cooling Systems (ECCS) sump in response to 2004 Generic Letter (GL) 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors.'' The changes add a license condition which requires that (1) Catawba Nuclear Station, Unit 1 will enter Mode 5 for the outage to install the sump strainer modification no later than May 19, 2008, and that (2) the Unit 1 sump strainer modification will be completed prior to entry into Mode 4 after May 19, 2008.

Date of issuance: October 31, 2007.

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

Amendment No.: 237.

Facility Operating License Nos. NPF–35: Amendment revises the license.

*Date of initial notice in **Federal Register:*** March 13, 2007 (72 FR 11386)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2007.

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: January 4, 2007.

Brief description of amendment: The amendment revised Technical Specifications (TSs) for the Limiting Conditions for Operation and Surveillance Requirements for Control Rod Operability, Scram Insertion Times, and Control Rod Accumulators.

Date of issuance: November 5, 2007.

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

Amendment No.: 230.

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

*Date of initial notice in **Federal Register:*** April 24, 2007 (72 FR 20381).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 2007.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois

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

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

Exelon Generation Company, LLC, Docket No. 50–352 and No. 50–353, Limerick Generating Station, Unit 1 and 2, Montgomery County, Pennsylvania

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: April 12, 2007.

Brief description of amendments: The amendments modify technical specification (TS) requirements related to control room envelope habitability in accordance with TS Task Force (TSTF) Traveler TSTF–448, Revision 2, ''Control Room Habitability.''

Date of issuance: October 31, 2007.

Effective date: As of the date of issuance, to be implemented within 180 days.

Amendment Nos.: 150, 150, 145, 145, 178, 186, 173, 188, 149, 264, and 268.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72, NPF–77, NPF–62, NPF–11, NPF–18, NPF–39, NPF–85, DPR–44, and DPR–56: The amendments revised the Technical Specifications and the Operating Licenses.

*Date of initial notice in **Federal Register:*** June 5, 2007 (72 FR 31100).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 10, 2007.

Brief description of amendment: The amendments revise the value of the safety limit minimum critical power ratio for the Dresden Nuclear Power Station (DNPS), Unit 2 technical specifications (TSs). The amendment also made conforming changes that clarify the wording of the DNPS, Unit 3 TSs.

Date of issuance: November 6, 2007.

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

Amendment Nos.: 224/216.

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

Date of initial notice in ***Federal Register***: July 31, 2007 (72 FR 41783), and September 5, 2007 (72 FR 50986).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 6, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 7, 2007, as supplemented by letter dated January 24, 2007.

Brief description of amendments: The amendments revise Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found main steam safety valve lift setpoint tolerance from ±1 percent to ±3 percent. In addition, the amendments revise TS SR 3.1.7.10 to increase the enrichment of sodium pentaborate used in the standby liquid control system from ≥30.0 atom percent boron-10 to ≥45.0 atom percent boron-10.

Date of issuance: November 1, 2007.

Effective date: As of the date of issuance and shall be implemented prior to main steam safety valve testing during the next refueling outage currently scheduled for May 2009 for Unit 1 and May 2008 for Unit 2.

Amendment Nos.: 235/230.

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

*Date of initial notice in **Federal Register***: January 30, 2007 (72 FR 4307) The January 24, 2007, 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 amendments is contained in a Safety Evaluation dated November 1, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 11, 2006.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7.7, ''Nuclear Services Closed Cycle Cooling Water (SW) System,'' to reduce the allowed outage time when one of the required SW heat exchangers is out of service.

Date of issuance: October 23, 2007.

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

Amendment No.: 225.

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

*Date of initial notice in **Federal Register***: February 13, 2007 (72 FR 6783).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 8, 2007, as supplemented by letter dated August 23, 2007.

Brief description of amendment: The amendment changes the basis for protection of the spent fuel stored in the spent fuel pool (SFP) in order to eliminate the Final Safety Analysis Report commitment for maintaining the SFP missile shields.

Date of issuance: October 24, 2007.

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

Amendment No.: 226.

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

*Date of initial notice in **Federal Register***: March 13, 2007 (72 FR 11381). The supplement dated August 23, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 5, 2006, as supplemented by letters dated April 4 and July 19, 2007.

Brief description of amendment: The amendment changes the restrictions on fuel storage in the spent fuel pool.

Date of issuance: October 25, 2007.

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

Amendment No.: 227.

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

*Date of initial notice in **Federal Register***: November 21, 2006 (71 FR 67394). The supplements dated April 4 and July 19, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 22, 2007.

Brief description of amendments: Amendments delete Section 3.H of Facility Operating License Nos. DPR–67 and NPF–16, which require reporting of violations of the requirements of Sections 3.A, 3.D, 3.F and 3.G of the operating license.

Date of Issuance: October 31, 2007.

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

Amendment Nos.: 203 and 150.

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

*Date of initial notice in **Federal Register***: June 19, 2007 (72 FR 33783).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: May 4, 2007.

Brief description of amendments: The proposed amendment would incorporate the administrative changes to Technical Specification (TS) 6.2.1.a, ''On and Offsite Organization'' and 6.8.1.a, ''Procedures and Programs.''

Date of issuance: November 2, 2007.

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

Amendment Nos: 236 and 231.

Renewed Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the TSs.

*Date of initial notice in **Federal Register***: July 3, 2007 (72 FR 36522).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 23, 2007.

Brief description of amendment: The amendment modifies Technical Specification 3.3.2.1, ''Control Rod Block Instrumentation,'' to allow a new banked position withdrawal sequence for shutdown, using the Consolidated Line Item Improvement Process.

Date of issuance: October 26, 2007.

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

Amendment No.: 120.

Renewed Facility Operating License No. NPF–69: Amendment revised the License and Technical Specifications.

*Date of initial notice in **Federal Register***: September 25, 2007 (72 FR 54477).

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated October 26, 2007.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: May 10, 2007.

Brief description of amendments: The requested changes are a partial adoption of Technical Specification Task Force (TSTF)–491, Revision 2, ''Removal of Main Steam and Feedwater Valve Isolation Times'' which was proposed by the TSTF by letter on May 18, 2006. The proposed changes revise Technical Specification (TS) 3.7.2 ''Main Steam Valves Closure Times'' by relocating the isolation valve closure times to a licensee-controlled document identified as a Bases reference. The proposed amendments deviate from TSTF–491 in that the current PINGP TS (3.7.3) and associated surveillance requirements for the main feedwater isolation valves do not include valve closure times, and thus, the changes to TS 3.7.3 provided for in TSTF–491 are not applicable to the PINGP TSs and are not adopted. TSTF change traveler TSTF–491, Revision 2, was announced for availability in the **Federal Register** on December 29, 2006, as part of the consolidated line item improvement process.

Date of issuance: October 31, 2007.

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

Amendment Nos.: 181 and 171.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: July 17, 2007 (72 FR 39083).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 17, 2007.

Brief description of amendments: The amendment modified Technical Specifications requirements related to control room envelope habitability in accordance with Technical Specifications Task Force 448, Revision 3, using the Consolidated Line Item Improvement Process.

Date of issuance: October 31, 2007.

Effective date: as of its date of issuance, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2–214; Unit 3–206.

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

Date of initial notice in ***Federal Register:*** May 22, 2007 (72 FR 28722). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: May 21, 2007, as supplemented by letter dated June 11, 2007.

Brief description of amendment: The amendment modified the technical specification (TS) requirements for inoperable snubbers by adding Limited Condition for Operation 3.0.8, using the Consolidated Line Item Improvement Process. The change is based on TS Task Force (TSTF) TSTF–372, Revision 4.

Date of issuance: October 17, 2007.

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

Amendment Nos.: 251, 231.

Renewed Facility Operating License Nos. NPF–4 and NPF–7: Amendments change the licenses and the technical specifications.

Date of initial notice in ***Federal Register***: June 19, 2007 (72 FR 33785) The supplement dated July 11, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 17, 2007.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: May 29, 2007.

Brief description of amendment: The amendments modify the Technical Specification requirements related to control room habitability, using the Technical Specification Task Force traveler, TSTF–448, revision 3.

Date of issuance: October 31, 2007.

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

Amendment Nos.: 252, 232.

Renewed Facility Operating License Nos. NPF–4 and NPF–7: Amendments change the licenses and the technical specifications.

Date of initial notice in ***Federal Register:*** July 3, 2007 (72 FR 36523).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 8th day of November 2007.

For The Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E7–22331 Filed 11–19–07; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–56784; File No. SR-CHX–2007–25]

Self-Regulatory Organizations; Chicago Stock Exchange, Inc.; Notice of Filing and Immediate Effectiveness of Proposed Rule Change as Modified by Amendment No. 1 Thereto to Eliminate References to the ITS Plan and Other Now-Obsolete Matters

November 14, 2007.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 (the ''Act''), [1] and Rule 19b–4 thereunder, [2] notice is hereby given that on October 17, 2007, the Chicago Stock Exchange, Inc. (''CHX'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission'') the proposed rule change as described in Items I and II below, which Items have been substantially prepared by the CHX. On November 9, 2007, CHX filed Amendment No. 1 to the proposed rule change. CHX has designated the proposed rule change as a ''non-controversial'' rule change pursuant to section 19(b)(3)(A) of the Act [3] and Rule

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] 15 U.S.C. 78s(b)(3)(A).