[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2013-0201]

Biweekly Notice

Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations

Background

Pursuant to Section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (NRC) 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 or combined license, as applicable, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 9, 2013, to August 21, 2013. The last biweekly notice was published on August 20, 2013 (78 FR 51219).

ADDRESSES: You may submit comment by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0201**. Address questions about NRC dockets to Carol Gallagher; telephone: 301-287-3422; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individual(s) listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: 3WFN, 06-44M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

For additional direction on accessing information and submitting comments, see "Accessing Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

SUPPLEMENTARY INFORMATION:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID **NRC-2013-0201** when contacting the NRC about the availability of information regarding this document. You may access publicly-available information related to this action by the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0201**.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly-available documents online in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. Documents may be viewed in ADAMS by performing a search on the document date and docket number.

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID **NRC-2013-0201** in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC posts all comment submissions at http://www.regulations.gov as well as entering the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined 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 § 50.92 of Title 10 of the *Code of Federal Regulations* (10 CFR), this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 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.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license or combined license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Agency Rules of Practice and Procedure" in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the NRC's PDR, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The NRC regulations are accessible electronically from the NRC Library on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/cfr/. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: 1) the name, address, and telephone number of the requestor or petitioner; 2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; 3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and 4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the requestor/petitioner 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 requestor/petitioner shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the requestor/petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. 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 requestor/petitioner to relief. A requestor/petitioner 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, 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, then any hearing held would take place before the issuance of any amendment.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory 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 an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at 301-415-1677, to request (1) a digital information (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 (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in the NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, Web-based submission form. In order to serve documents through the Electronic Information Exchange System, users will be required to install a Web browser plug-in from the NRC's Web site. Further information on the Web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with the NRC's guidance available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing 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 E-Filing system also distributes an e-mail notice that provides access to the document to the NRC's 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 using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC's Meta System Help Desk through the "Contact Us" link located on the NRC's Web site at http://www.nrc.gov/site-help/e-submittals.html, by e-mail to MSHD.Resource@nrc.gov, or by a toll-free call at 1-866 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, 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. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at http://ehd1.nrc.gov/ehd/, unless excluded pursuant to an order of the Commission, or the 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, unless an NRC regulation or other law requires submission of such information. However, a request to intervene will require including information on local residence in order to demonstrate a proximity assertion of interest in the proceeding. 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.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Requests for hearing, petitions for leave to intervene, and motions for leave to file new or amended contentions that are filed after the 60-day deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i)-(iii).

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection at the NRC's PDR, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through ADAMS in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC's PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

Exelon Generation Company (EGC), LLC, Docket Nos. 50-373, and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: October 15, 2012, and August 12, 2013.

Description of amendment request: The proposed amendments would remove License Conditions which are no longer necessary to address an interim configuration of the LaSalle County Station (LSCS), Unit 2, spent fuel pool prior to completing installation of NETCO-SNAP-IN® inserts. By letter dated August 12, 2013, EGC provided additional information and expanded the scope of the application as originally noticed. The August 12, 2013, letter proposed to clarify language in the LSCS, Units 1 and 2, Technical Specifications (TS) applicable to the design features for TS 4.3, 'Fuel Storage.' The proposed amendment was initially published in the *Federal Register* Biweekly notice on April 2, 2013 (78 FR 19751).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided on August 12, 2013, its revised 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 removes License Conditions within the LSCS Unit 2 Operating License related to interim configurations of the SFP during the installation of the NETCO-SNAP-IN® inserts and the required completion date for installation. The proposed change also revises TS Section 4.3.1 to clarify that for the Unit 2 SFP, spent fuel shall only be stored in storage rack cells containing a neutron absorbing rack insert. All changes proposed by EGC in this license amendment request are administrative in nature because they remove License Conditions that have either been satisfied or that are no longer applicable, and the revision to TS Section 4.3.1 ensures spent fuel is stored only in cells that contain inserts. There are no physical changes to the facilities, nor any changes to the station operating procedures, limiting conditions for operation, or limiting safety system settings.

   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 removes License Conditions within the LSCS Unit 2 Operating License related to interim configurations of the SFP during the installation of the NETCO-SNAP-IN® inserts and the required completion date for installation. The proposed change also revises TS Section 4.3.1 to clarify that for the Unit 2 SFP, spent fuel shall only be stored in storage rack cells containing a neutron absorbing rack insert. There are no changes to the SFP criticality analysis associated with the proposed change. No physical changes to the plant are proposed, and there are no changes to the manner in which the plant is operated. Rather, the proposed change is administrative because it involves removing License Conditions that have either been satisfied or that are no longer applicable, and the revision to TS Section 4.3.1 ensures spent fuel is stored only in cells that contain inserts.

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

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

Response: No.

The proposed change removes License Conditions within the LSCS Unit 2 Operating License related to interim configurations of the SFP during the installation of the NETCO-SNAP-IN® inserts and the required completion date for installation. The proposed change also revises TS Section 4.3.1 to clarify that for the Unit 2 SFP, spent fuel shall only be stored in storage rack cells containing a neutron absorbing rack insert. Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in Technical Specifications. The proposed change does not alter these established safety margins. The proposed change does not alter the criticality analysis for the SFP and does not affect the SFP criticality safety margin. The proposed change is administrative because it involves removing License Conditions that have either been satisfied or that are no longer applicable, and the revision to TS Section 4.3.1 ensures spent fuel is stored only in cells that contain inserts.

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

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

Attorney for licensee: Ms. Tamra Domeyer, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

Acting NRC Branch Chief: Jeremy S. Bowen.

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 amendment request: June 10, 2013.

Description of amendment request: The proposed amendment revises Technical Specification (TS) Surveillance Requirements (SR) 3.8.4.2 and 3.8.4.5. The proposed change would resolve a non-cited violation (NCV) that was documented in an NRC's Inspection Report. Specifically, the NRC identified an NCV for the failure to verify that safety-related batteries would remain operable if all the inter-cell and terminal connections were at the maximum resistance value allowed by SR 3.8.4.2 and SR 3.8.4.5 (i.e., 150 micro-ohms). The proposed change maintains the existing resistance limit for inter-cell and terminal connections, and adds new acceptance criteria for total battery connection resistance to ensure that the safety-related batteries can perform their specified safety function.

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 revisions of SR 3.8.4.2 and SR 3.8.4.5 to add a battery connector resistance acceptance criterion will not challenge the ability of the safety·related batteries to perform their safety function. The total battery connection resistance is a parameter that is representative of overall battery performance, and ensures that the safety-related batteries remain capable of performing their specified safety function. Appropriate monitoring and maintenance will continue to be performed on the safety-related batteries. In addition, the safety-related batteries are within the scope of 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” which will ensure the control of maintenance activities associated with this equipment.

Current TS requirements will not be altered and will continue to require that the equipment be regularly monitored and tested. Since the proposed change does not alter the manner in which the batteries are operated, there is no significant impact on reactor operation.

The proposed change does not involve a physical change to the batteries, nor does it change the safety function of the batteries. The DC power system/batteries will retain adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components.

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

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

   Response: No.

   The proposed changes revising SR 3.8.4.2 and SR 3.8.4.5 to add an additional acceptance criterion for battery connector resistance is an increase in conservatism, without a change in system testing methods, operation, or control. Safety-related batteries installed in the plant will be required to meet criteria more restrictive and conservative than current acceptance criteria and standards. The proposed change does not affect the manner in which the batteries are tested and maintained; therefore, there are no new failure mechanisms for the system.

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

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

Response: No.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not modify the safety limits or setpoints at which protective actions are initiated. The new acceptance criterion is more restrictive than the existing acceptance criteria for inter-cell and terminal connection resistance, and the proposed change ensures the availability and operability of safety-related battery operability and availability.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

Acting NRC Branch Chief: Jeremy Bowen.

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

Date of amendment request: July 5, 2013.

Description of amendment request: The proposed amendment includes supporting changes to NMP2 Technical Specification (TS) 3.1.7, “Standby Liquid Control (SLC) System,” to increase the isotopic enrichment of boron-10 in the sodium pentaborate solution utilized in the SLC System and decrease the SLC System tank volume. The following are the proposed changes to the NMP2 TS 3.1.7, “Standby Liquid Control (SLC) System”:

- Revise the acceptance criterion in SR 3.1.7.10 by increasing the sodium pentaborate boron-10 enrichment requirement from ≥ 25 atom percent to ≥ 92 atom percent, and make a corresponding change in TS Figure 3.1.7-1, "Sodium Pentaborate Solution Volume/Concentration Requirements."
- Revise TS Figure 3.1.7-1 to account for the decrease in the minimum volume of the SLC system tank. At a sodium pentaborate concentration of 13.6% the minimum volume changes from 4,558.6 gallons to 1,600 gallons. At a sodium pentaborate concentration of 14.4%, the minimum volume changes from 4,288 gallons to 1,530 gallons.

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

   Response: No.

   The SLC System is used to mitigate the consequences of an Anticipated Transient Without SCRAM (ATWS) special event and is used to limit the radiological dose during a Loss of Coolant Accident (LOCA). The proposed changes do not affect the capability of the SLC System to perform these two functions in accordance with the assumptions of the associated analyses.

   A SLC System failure is not a precursor of any previously evaluated accident in the NMP2 Updated Safety Analysis Report (USAR). Consequently there is no change in the probability of an accident previously evaluated.

   The current ATWS analysis is not adversely affected by the proposed changes because the reactivity insertion rate would increase by a factor greater than 3 and the amount of injected boron-10 is not reduced. The ability of the SLC System to mitigate radiological dose in the event of a LOCA is not affected by these changes.

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

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

   Response: No.

   Structures, systems and components (SSCs) previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes do not adversely affect safety-related SSCs and do not challenge the performance or integrity of any safety-related SSC. The physical changes to the SLC System are limited to the increase in the boron-10 enrichment of the sodium pentaborate solution in the SLC System storage tank, the corresponding decrease in the net sodium pentaborate solution volume requirement in the SLC System storage tank, and the associated instrumentation changes. In addition, the effective SLC System flow rate utilized in the boron equivalency analysis is reduced. The proposed changes do not otherwise affect the design or operation of the SLC System.

   This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than was previously evaluated.

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

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

   Response: No.

   The SLC System is used to mitigate the consequences of an ATWS event and is used to limit the radiological dose during a LOCA. The proposed changes do not affect the capability of the SLC System to perform these two functions in accordance with the assumptions of the associated analyses. The current ATWS analysis is not adversely affected by the proposed changes because the reactivity insertion rate would increase by a factor greater than 3 and the amount of injected boron-10 is not reduced. The ability of the SLC System to mitigate radiological dose in the event of a LOCA by maintaining suppression pool pH ≥ 7.0 is not affected by these changes.

   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: Gautam Sen, Senior Counsel, Constellation Energy Nuclear Group, LLC, 100 Constellation Way, Suite 200C, Baltimore, MD 21202.

Acting NRC Branch Chief: Robert Beall.

Northern States Power Company - Minnesota, Docket No. 50-263, Monticello Nuclear Generating Plant (MNGP), Wright County, Minnesota

Date of amendment request: April 19, 2013.

Description of amendment request: The licensee proposed to revise MNGP Technical Specification (TS) 1.1, “Definitions,” to modify the definition of “Shutdown Margin (SDM)” to require calculation of the SDM at a reactor moderator temperature of 68 degrees Fahrenheit (°F), or at a higher temperature that represents the most reactive state throughout the operating cycle. This change is needed for newer boiling water reactor fuel designs which may be more reactive at shutdown temperatures above 68°F. The proposed change is consistent with Technical Specifications Task Force (TSTF) Traveler TSTF-535, Revision 0, “Revise Shutdown Margin Definition to Address Advanced Fuel Designs.” Notice of availability of TSTF-535 was published in the *Federal Register* on February 26, 2013 (78 FR 13100).

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

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

   Response: No.

The proposed change revises the definition of SDM. SDM is not an initiator to any accident previously evaluated. Accordingly, the proposed change to the definition of ADM has no effect on the probability of any accident previously evaluated. ADM is an assumption in the analysis of some previously evaluated accidents and inadequate SDM could lead to an increase in consequences for those accidents. However, the proposed change revised the SDM definition to ensure that the correct SDM is determined for all fuel types at all times during the fuel cycle. As a result, the proposed change does not adversely affect the consequences of any accident previously evaluated.

Therefore, it is concluded that these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change revises the definition of SDM. The change does not involve a physical alteration of the plant (i.e., no new of different type of equipment will be installed) or a change in methods governing normal plant operations. The change does not alter assumptions made in the safety analysis regarding SDM.

   Therefore, it is concluded that these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed change revised the definition of SDM. 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 ensures that the SDM assumed in determining safety limits, limiting safety system settings or limiting conditions for operation is correct for all BWR fuel types at all times during the fuel cycle.

   Therefore, it is concluded that these 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 the licensee: Peter M. Glass, Assistant General Counsel, Xcel Energy Services, Inc., 414 Nicollet Mall, Minneapolis, MN 55401.

NRC Branch Chief: Robert D. Carlson.

Pacific Gas and Electric Co., Docket No. 50-133, Humboldt Bay Power Plant (HBPP), Unit 3 Humboldt County, California

Date of amendment request: May 3, 2013.

Description of amendment request: The proposed amendment would add License Condition 2.C.5 that approves the License Termination Plan (LTP) and adds a license condition that establishes the criteria for determining when changes to the LTP require prior NRC approval.

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

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

   Response: No.

   The change allows for the approval of the LTP and provides the criteria for when changes to the LTP require prior NRC approval. This change does not affect possible initiating events for the decommissioning accidents previously evaluated in the Humboldt Bay Power Plant (HBPP) defueled safety analysis report (DSAR), as updated, appendix A, "Implications of Decommissioning Accidents with Potential for Radiological Impacts to the Environment,' or alter the configuration or operation of the facility. Safety limits, limiting safety system settings, and limiting control systems are no longer applicable to HBPP in the permanently defueled mode, and are therefore not relevant.

The proposed change does not affect the boundaries used to evaluate compliance with liquid or gaseous effluent limits, and has no impact on plant operations.

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

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

   Response: No.

   The safety analysis for the facility remains accurate as described in the HBPP DSAR, as updated, appendix A. There are sections of the LTP that refer to the decommissioning activities still remaining (e.g. removal of large components, decontamination, etc.). However, these activities are performed in accordance with approved HBPP work packages/steps and undergo 10 CFR 50.59 screening prior to initiation. The proposed amendment merely makes mention of these processes and does not bring about physical changes to the facility. Therefore, the facility conditions for which the postulated accidents have been evaluated are still valid and no new accident scenarios, failure mechanisms, or single failures are introduced by this amendment. The system operating procedures are not affected.

   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 change involve a significant reduction in a margin of safety?

   Response: No.

   There are no changes to the design or operation of the facility resulting from this amendment. The proposed change does not affect the boundaries used to evaluate compliance with liquid or gaseous effluent limits, and has no impact on plant shutdown operations. Accordingly, neither the postulated accident assumptions in the DSAR, as updated, appendix A, nor the Technical Specifications are affected.

   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. Jennifer K. Post, Pacific Gas and Electric Company, 77 Beale Street, B30A, San Francisco, CA.

NRC Branch Chief: Bruce Watson.

South Carolina Electric and Gas, Docket Nos.: 52-027 and 52-028, Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, Fairfield County, South Carolina

Date of amendment request: July 17, 2013.

Description of amendment request: The proposed amendment would depart from VCSNS Units 2 and 3 plant-specific Design Control Document (DCD) Tier 2 and Tier 2* material contained within the Updated Final Safety Analysis Report (UFSAR) to acknowledge various obstructions and interferences (other than wall openings and penetrations) that may cause a change to the design spacing of shear studs and the design and spacing of wall module trusses in a local area, and to acknowledge appropriate weld types.

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 design function of the containment structural modules is to support the reactor coolant system components and related piping systems and equipment. The design functions of the affected structural modules in the auxiliary building are to provide support and protection for new and spent fuel and the equipment needed to support fuel handling, cooling, and storage in the spent fuel racks, and to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located outside the containment building. The design function of the shear studs is to enable the concrete and steel faceplates to act in a composite manner and transfer loads into the concrete of the structural modules. The structural modules are seismic Category I structures and are designed for dead, live, thermal, pressure, safe shutdown earthquake loads, and loads due to postulated pipe breaks. The loads and load combinations applicable to the structural modules in the auxiliary building are the same as for the containment internal structures except that there are no design basis accident loadings due to the automatic depressurization system or pressure loads due to pipe breaks. The proposed changes to the UFSAR are to include types of interferences other than wall openings and penetrations that may cause a change in the design spacing of shear studs and the design and spacing of wall module trusses in a local area. The proposed changes clarify that the stud spacing is specified as a design value and add the tolerance for stud spacing. The revised spacing including the tolerance continues to be in conformance with the design and analysis requirements identified in the UFSAR. The proposed changes also include clarification of a requirement for a complete joint penetration weld. The thickness, geometry, and strength of the structures are not adversely altered. The material of the steel plates is not altered. The properties of the concrete included in the structural modules are not altered. As a result, the design function of the containment structural modules is not adversely affected by the proposed change. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor does the change described create any new accident precursors.

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 changes to the UFSAR acknowledge types of interferences (other than wall openings and penetrations) that may cause a change in the typical design spacing of shear studs and the design and spacing of wall module trusses in a local area. The proposed changes clarify that the stud spacing is specified as a design value and provide the tolerance for stud spacing. The revised spacing, including the tolerance, continues to be in conformance with the design and analysis requirements identified in the UFSAR. Stud spacing and sizing are evaluated to demonstrate that stud loadings and shear transfer capability are within acceptable limits and that the structural module acts in a composite manner. An additional proposed change is to clarify a requirement for a complete joint penetration weld. The thickness, geometry, and strength of the structures are not adversely altered. The materials of the steel plates are not altered. The properties of the concrete included in the structural modules are not altered. The changes to the internal design of the structural modules do not create any new accident precursors. As a result, the design function of the modules is not adversely affected by the proposed changes.

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

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

Response: No.

The criteria and requirements of American Concrete Institute (ACI) 349 and American Institute of Steel Construction (AISC) N690 provide a margin of safety to structural failure. The design of the shear studs and wall trusses for the structural wall modules conforms to applicable criteria and requirements in ACI 349 and AISC N690 and, therefore, maintain the margin of safety. The proposed changes to the UFSAR acknowledge types of interferences (other than wall openings and penetrations) that may cause a change in the typical design spacing of shear studs and the design and spacing of wall module trusses in a local area. The proposed changes clarify that the stud spacing is specified as a design value and add the tolerance for stud spacing. The revised spacing including the tolerance continues to be in conformance with the design and analysis requirements identified in the UFSAR. An additional proposed change is to clarify a requirement for a complete joint penetration weld. There is no change to the capacity of the weld or to the design requirements of the modules. There is no change to the method of evaluation from that used in the design basis calculations.

Therefore, the proposed amendment does not result in a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Kathryn M. Sutton, Morgan, Lewis & Bockius LLC, 1111 Pennsylvania Avenue, NW, Washington, DC 20004-2514.

NRC Branch Chief: Lawrence Burkhart.

Southern Nuclear Operating Company, Docket Nos. 52-025, and 52-026, Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Burke County, Georgia

Date of amendment request: March 15, 2013, and revised on July 10, 2013, and supplemented on August 16, 2013.

Description of amendment request: The proposed change would amend Combined License Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 by departing from the plant-specific Design Control Document (DCD) Tier 1(and corresponding Combined License Appendix C information) and Tier 2 material by making changes to the Non-Class 1E dc and Uninterruptible Power Supply System (EDS) and Uninterruptible Power Supply System (IDS) and making changes to the corresponding Tier 1 information in Appendix C to the Combined License. The proposed changes would:

(1) Increase EDS total equipment capacity, component ratings, and protective device sizing to support increased load demand,

(2) Relocate equipment and moving Turbine Building (TB) first bay EDS Battery Room and Charger Room. The floor elevation increases from elevation 148’-0” to elevation 148’-10” to accommodate associated equipment cabling with this activity, and

(3) Remove the Class 1E IDS Battery Back-up tie to the Non-Class 1E EDS Battery.

Because this proposed change requires a departure from Tier 1 information in the Westinghouse Advanced Passive 1000 design control document (DCD), the licensee also requested an exemption from the requirements of the Generic DCD Tier 1 in accordance with 52.63(b)(1).

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

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

   Response: No.

   The design function of the Turbine Building (TB) is to provide weather protection for the laydown and maintenance of major turbine/generator components. The TB first bay is a seismic Category II structure designed to prevent the collapse under a safe shutdown earthquake (SSE) to protect the adjacent auxiliary building. The electrical system and air-handling units are designed to provide electrical power to plant loads and maintain acceptable temperatures for electrical equipment rooms and work areas. The electrical equipment continues to be in accordance with the same codes and standards stated in the Updated Final Safety Analysis Report (UFSAR). The proposed relocation of equipment, including the increase in floor elevation by 10 inches to accommodate overhead equipment cabling, does not impact the TB design function. The TB first bay continues to meet seismic Category II requirements. Based on this, the proposed changes would not increase the probability of an accident previously evaluated.

   The proposed changes do not involve any accident initiating event, thus the probabilities of the accidents previously evaluated are not affected. The relocation of equipment does not involve any safety-related structures, systems, or components; the affected rooms do not represent a radioactive material barrier; and this activity does not affect the containment of radioactive material. The radioactive material source terms and release paths used in the safety analyses are unchanged, thus the radiological releases in the accident analyses are not affected. Therefore, the consequences of an accident previously evaluated are not affected.

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

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

   Response: No.

   The proposed changes would use the same type of electrical equipment with higher ratings and capacity, change the source of a battery back-up, and relocate equipment. The electrical equipment will continue to perform its design functions because the same electrical codes and standards as stated in the UFSAR continue to be met. Therefore the proposed changes do not affect equipment failure probabilities or alter any accident initiator or initiating sequence of events. The proposed changes in location of equipment and elevation of the TB first bay floor do not affect the design function of the TB first bay to protect the adjacent auxiliary building by meeting seismic Category II structure requirements, or affect the operation of the relocated equipment, or the ability of the relocated equipment to meet its design functions. Because the SSCs and equipment affected by the proposed changes continue to meet their design functions, the structural codes and standards as stated in the UFSAR, the proposed changes do not introduce a different type of accident than those previously considered.

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

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

   Response: No.

   The current seismic requirements applicable to the seismic Category II TB first bay structure, including the seismic modeling and analysis methods, will continue to apply to the TB first bay floor elevation increase. The proposed changes to relocate equipment and the increase in the floor elevation will continue to meet the fire rating requirements and will be in accordance with the same codes and standards currently identified in the UFSAR. The proposed changes to the electrical equipment will continue to meet existing electrical equipment industry standard recommendations identified in the UFSAR. Because no safety analysis or design basis acceptance limit/criterion is challenged or exceeded by these proposed changes, no margin of safety is reduced.

   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: Mr. M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Lawrence Burkhart.

Southern Nuclear Operating Company, Docket Nos. 52-025, and 52-026, Vogtle Electric Generating Station (VEGP) Units 3 and 4, Burke County, Georgia

Date of amendment request: July 2, 2013.

Description of amendment request: The proposed change would amend Combined License Nos. NPF-91, and NPF-92 for VEGP Units 3 and 4, respectively, by revising Tier 2* and associated Tier 2 information related to the design details of connections in several locations between the steel plate composite construction (SC) used for the shield building and the standard reinforced concrete (RC) walls, floors, and roofs of the auxiliary building and lower walls of the shield building. These connections are also referred to as “RC to SC connections.”

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 design functions of the nuclear island structures are to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in the nuclear island. The nuclear island structures are structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29. The change to the detail design of connections between the RC and SC structures do not have an adverse impact on the response of the nuclear island structures to safe shutdown earthquake ground motions or loads due to anticipated transients or postulated accident conditions. The changes to the detail design do not impact the support, design, or operation of mechanical and fluid systems. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor do the changes describe create any new accident precursors.

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 changes are to the detail design of connections between the RC and SC structures. The changes to the detail design of connections do not change the criteria and requirements for the design and analysis of the nuclear island structures. The changes to the detail design of connections do not change the design function, support, design, or operation of mechanical and fluid systems. The changes to the detail design of connections do not change the methods used to connect the RC to the SC. The changes to the detail design of the connections do not result in a new failure mechanism for the nuclear island structures or new accident precursors. As a result, the design functions of the nuclear island structures are not adversely affected by the proposed changes.

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

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

Response: No.

No safety analysis or design basis acceptance limit/criterion is involved by the requested changes, thus, no margin of safety is reduced.

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

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

Attorney for licensee: Mr. M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Lawrence J. Burkhart.

Southern Nuclear Operating Company, Docket Nos. 52-025, and 52-026, Vogtle Electric Generating Station (VEGP) Units 3 and 4, Burke County, Georgia

Date of amendment request: July 15, 2013.

Description of amendment request: The proposed change would amend Combined License Nos. NPF-91, and NPF-92 for VEGP Units 3 and 4, respectively, by revising Tier 2* information related to the construction of Module CA03. Some of these changes include the removal of specifically mentioned materials, increasing anchoring supports and allowing the use of anchor bars with hooks.

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 design functions of the nuclear island structures are to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in the nuclear island. The nuclear island structures are structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29. The change to the design details for the in-containment refueling water storage tank (IRWST) west wall does not have an adverse impact on the response of the nuclear island structures to safe shutdown earthquake ground motions or loads due to anticipated transients or postulated accident conditions, nor does it change the seismic Category I classification. The change to the design details for the IRWST west wall does not impact the support, design, or operation of mechanical and fluid systems. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor does the change described create any new accident precursors.

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 change is to revise design details for the IRWST west wall. The change of the design details for the IRWST west wall does not change the design requirements of the nuclear island structures, nor the seismic Category I classification. The change of the design details for the IRWST west wall does not change the design function, support, design, or operation of mechanical and fluid systems. The change of the design details for the IRWST west wall does not result in a new failure mechanism for the nuclear island structures or introduce any new accident precursors. As a result, the design function of the nuclear island structures is not adversely affected by the proposed change.

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

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

   Response: No.

   No safety analysis or design basis acceptance limit/criterion is involved by the requested changes, thus, no margin of safety is reduced.

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

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

Attorney for licensee: Mr. M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Lawrence J. Burkhart.

Southern Nuclear Operating Company, Docket Nos. 52-025, and 52-026, Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Burke County, Georgia

Date of amendment request: August 6, 2013.

Description of amendment request: The proposed change would amend Combined License Nos. NPF-91, and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 by departing from the plant-specific Design Control Document (DCD) Tier 1(and corresponding Combined License Appendix C information) and Tier 2 material by revising the safety function and classification of Liquid Radwaste System (WLS) drain hubs in the Chemical and Volume Control System and Passive Core Cooling System (PXS) compartments. In addition, the proposed changes would modify the PXS compartment drain piping connection; WLS valve types, and depiction of components in the WLS figures.

Because, this proposed change requires a departure from Tier 1 information in the Westinghouse Advanced Passive 1000 DCD, the licensee also requested an exemption from the requirements of the Generic DCD Tier 1 in accordance with 52.63(b)(1).

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

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

   Response: No.

   The design function of the WLS is containment isolation and the prevention of backflow in the drain lines from the CVS compartment and the PXS compartment to the containment sump which prevents cross flooding of these compartments. The proposed changes to the WLS drainage function; the CVS and PXS compartment drain hubs; and the WLS valve types do not affect these design functions or any other system design function. Revising the drain hub safety classification, the PXS drains connection type, and the WLS valve types do not involve any accident initiating event or component failure. The changes to how components (valves, filters) are depicted in the figure provide consistency with the figure legend and do not alter any system functions. The system will utilize the same codes and standards previously used for the system. Since there are no impacts on accident initiating events or component failures, the probability of an accident previously evaluated is not affected. The radioactive material source terms and release paths used in the safety analyses are unchanged, thus the radiological releases in the Updated Final Safety Analysis Report (UFSAR) accident analyses are not affected.

   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 changes to the WLS system do not adversely affect the design or quality of any structure, system or component. Revising the WLS safety functions and re-classifying the drain hubs as nonsafety-related does not create a new fault or sequence of events that could result in a radioactive material release nor do the changes to the WLS piping connections, valve types and the depiction of components on the figure have any impact on any accident previously evaluated.

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

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

Response: No.

The proposed changes to the WLS system drain hubs, piping connection, valve type, and Tier 1 figure depiction would not affect any radioactive material barrier. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed change, thus no margin of safety is reduced.

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

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

Attorney for licensee: Mr. M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Lawrence J. Burkhart.

Southern Nuclear Operating Company, Inc., Docket Nos. 50-348, and 50-364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: December 21, 2012, as supplemented on May 21, 2013.

Description of amendment request: The proposed amendments would revise the Joseph M. Farley Nuclear Plant (FNP) Facility Operating Licenses (FOL), Appendix C, to require Southern Nuclear Operating Company (SNC) to fully implement and maintain in effect the Degraded Voltage Protection modification schedule.

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

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

   Response: No.

   The proposed change to the FNP FOL that incorporates the Degraded Voltage Protection modification implementation schedule is administrative in nature. This proposed change does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested or inspected.

   Therefore, this 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 to the FNP FOL that incorporates the Degraded Voltage Protection modification implementation schedule is administrative in nature. This proposed change does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested or inspected.

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

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

   Response: No.

   Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications. The proposed change to the FNP FOL is administrative in nature. Because there is no change to these established safety margins as a result of this change, the proposed change does not involve a significant reduction in a margin of safety.

> Therefore, the proposed change does not involve a significant reduction in 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: Leigh D. Perry, SVP & General Counsel, Southern Nuclear Operating Company, 40 Inverness Center Parkway, Birmingham, AL 35242.

NRC Branch Chief: Robert J. Pascarelli.

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: December 21, 2012, as supplemented June 21, 2013.

Description of amendment request: The proposed License Amendment Request (LAR) would revise the Edwin I. Hatch Nuclear Plant (HNP) Facility Operating Licenses to require Southern Nuclear Operating Company (SNC) to implement modifications that will eliminate the need for administrative controls with regard to protection of the plant from degraded grid voltage conditions for HNP.

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

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

   Response: No.

The proposed change to the HNP FOL that incorporates the Degraded Voltage Protection modification implementation schedule is administrative in nature. This proposed change does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested or inspected.

Therefore, this 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 to the HNP FOL that incorporates the Degraded Voltage Protection modification implementation schedule is administrative in nature. This proposed change does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested or inspected.

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

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

Response: No.

Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications. The proposed change to the HNP FOL is administrative in nature. Because there is no change to these established safety margins as a result of this change, the proposed change does not involve a significant reduction in a margin of safety.

Therefore, the proposed change does not involve a significant reduction in 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: Leigh D. Perry, SVP & General Counsel, Southern Nuclear Operating Company, 40 Inverness Center Parkway, Birmingham, AL 35242.

NRC Branch Chief: Robert J. Pascarelli.

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 (HNP), Units 1 and 2, Appling County, Georgia

Date of amendment request: July 23, 2013.

Description of amendment request: The proposed amendments would modify Technical Specification (TS) requirements related to control room envelope (CRE) habitability in accordance with the Nuclear Regulatory Commission (NRC)-approved Revision 3 of Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-448, "Control Room Habitability."

The NRC staff published a notice of opportunity for comment in the *Federal Register* on October 17, 2006 (71 FR 61075), on possible license amendments adopting TSTF-448 using the NRC's consolidated line-item improvement process (CLIIP) for amending licensees' TSs, which included a model safety evaluation (SE) and model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the *Federal Register* on January 17, 2007 (72 FR 2022), which included the resolution of public comments on the model SE and model NSHC determination. The licensee affirmed the applicability of the following NSHC determination in its application dated July 23, 2013.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC 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 as 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 proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Leigh D. Perry, SVP & General Counsel, Southern Nuclear Operating Company, 40 Inverness Center Parkway, Birmingham, AL 35242.

NRC Branch Chief: Robert Pascarelli.

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

Date of amendment request: July 3, 2013 (SQN-TS-12-04).

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) 3/4.6.5, “Ice Condenser.” The proposed changes would revise TS Limiting Condition for Operation 3.6.5.1.d and TS Surveillance Requirement 4.6.5.1.d.2 to raise the overall ice condenser ice weight from 2,225,880 pounds (lbs) to 2,540,808 lbs and to raise the minimum TS ice basket weight from 1145 lbs to 1307 lbs, respectively. These changes are necessary to address the issues raised in Nuclear Safety Advisory Letter (NSAL) 11-5, “Westinghouse LOCA [Loss-of-Coolant Accident] Mass and Energy Release Calculation

Issues." The issues identified in NSAL-11-5 affected plant-specific LOCA mass and energy release calculation results that are used as input to the containment integrity response analyses. The basis for the proposed changes is provided in WCAP-12455, Revision 1, Supplement 2R, "Tennessee Valley Authority [TVA] Sequoyah Nuclear Plant [SQN] Units 1 and 2 Containment Integrity Reanalyses Engineering Report."

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 consequence of an accident previously evaluated?

   Response: No.

   The analyzed accidents of consideration in regards to changes affecting the ice condenser are a loss of coolant accident (LOCA) and a main steam line break (MSLB) inside containment. The ice condenser is a passive system and is not postulated as being the initiator of any LOCA or MSLB and is designed to remain functional following a design basis earthquake. In addition, the ice condenser does not interconnect or interact with any systems that have an interface with the reactor coolant or main steam systems.

   For SQN, the LOCA is the more severe accident in terms of containment pressure and ice bed melt out, and is therefore the more limiting accident. The revised SQN LOCA containment integrity analysis determined that the post-LOCA peak containment pressure is below the containment design pressure and that the margin to ice meltout is maintained. The analysis assumes an ice weight that ensures sufficient heat removal capability is available from the ice condenser to limit the accident peak pressure inside containment.

   TVA has evaluated the effects of the increased ice condenser ice weight and determined that the increase in ice weight does not invalidate the ice condenser seismic qualification, does not adversely affect the capacity of the ice bed to absorb iodine during a LOCA, and does not diminish the boron concentration of the recirculated primary coolant during a LOCA. TVA has also evaluated differences between the as-built plant and the assumptions of the revised analysis and determined that the results of the revised analysis remain valid for Model 57AG steam generators and for AREVA Advanced W17 High Thermal Performance (HTP) fuel.

The proposed changes reflect the ice weight assumed in the containment integrity analysis including conservative allowances for sublimation and weighing instrument systematic error. Accordingly, the proposed changes ensure that ice weight values maintain margin between the calculated peak containment accident pressure and the containment design pressure. The results of the analysis and the margins are maintained; therefore, the consequences of a previously evaluated accident are not adversely affected by the proposed changes.

Because 1) the ice condenser is not an accident initiator, 2) the results of the revised analysis remain valid for Model 57AG steam generators and for AREVA Advanced W17 High Thermal Performance (HTP) fuel, and 3) the proposed changes to the TSs are limited to revision of the ice weight values to reflect the revised containment integrity analysis, there is no change in the probability of an accident previously evaluated in the SQN Updated Final Safety Analysis Report (UFSAR).

Based on the above discussions, the proposed changes do not involve an 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 ice condenser serves to limit the peak pressure inside containment following a LOCA or MSLB. The proposed changes are limited to the revision of the minimum ice weights specified in the TSs. The revised containment pressure analysis determined that sufficient ice would be present to maintain the peak containment pressure below the containment design pressure. No new modes of operation, accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed change.

TVA has evaluated the effects of the increased ice condenser ice weight and determined that the increase in ice weight does not invalidate the ice condenser seismic qualification, does not adversely affect the capacity of the ice bed to absorb iodine during a LOCA, and does not diminish the boron concentration of the recirculated primary coolant during a LOCA. TVA has also evaluated differences between the as-built plant and the assumptions of the revised analysis and determined that the results of the revised analysis remain valid for Model 57AG steam generators and for AREVA Advanced W17 High Thermal Performance (HTP) fuel. Because sufficient ice weight is available to maintain the peak containment pressure below the containment design pressure, the results of the revised analysis remain valid for Model 57AG steam generators and for

AREVA Advanced W1 7 High Thermal Performance (HTP) fuel, and the increase in ice weight does not invalidate the ice condenser seismic qualification, the increased ice weight does not create the possibility of an accident that is different than any already evaluated in the SQN UFSAR.

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

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

Response: No.

The operability of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The revised analysis demonstrates that the ice condensers will continue to preclude over-pressurizing the lower containment and continue to absorb sufficient heat energy to assist in precluding containment vessel failure. TVA has evaluated the effects of the increased ice condenser ice weight and determined that the increase in ice weight does not invalidate the ice condenser seismic qualification, does not adversely affect the capacity of the ice bed to absorb iodine during a LOCA, and does not diminish the boron concentration of the recirculated primary coolant during a LOCA.

The proposed changes are required to resolve non-conservative TSs currently addressed by administrative controls established in accordance with Nuclear Regulatory Commission (NRC) Administrative Letter 98-10. The revised containment integrity response analysis requires an increase in the required ice weight to ensure that the post-LOCA peak containment pressure remains within the design limits. As a result, the proposed changes restore margin between the accident peak pressure and the containment design pressure and resolve non-conservative TSs ice weight values currently under administrative controls. Accordingly, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Acting NRC Branch Chief: Douglas A. Broaddus.

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

Virginia Electric and Power Company, Docket No. 50-280, and 50-281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of amendment request: June 26, 2013.

Description of amendment request: The proposed license amendment (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13179A014) requests the approval of (1) generic application of Appendix D, "Qualification of the ABB-NV and WLOP Critical Heat Flux (CHF) Correlations in the Dominion VIPRE-D Computer Code," to Fleet Report DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," (2) the plant-specific application of Appendix D to DOM-NAF-2-A to North Anna and Surry Power Stations (in accordance with Section 2.1 of DOM-NAF-2-A), and (3) an increase in the Surry Power Station Technical Specification Minimum Temperature for Criticality.

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

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

   The first and second proposed changes would allow Dominion to use the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs to perform licensing calculations for North Anna and Surry, using the DDLs documented in Appendix D of Fleet Report DOM-NAF-2. Neither code/correlation pair methodology makes any contribution to the potential accident initiators and thus cannot increase the probability of any accident. Further, since the DDLs for ABB-NV and WLOP meet the required design basis of avoiding departure from nucleate boiling (DNB) with 95% probability at a 95% confidence level, the use of the new code/correlations does not increase the potential consequences of any accident. The pertinent evaluations that need to be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable have been performed and determined to be acceptable. The use of a different code/correlation pair will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. The use of the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs to perform licensing calculations for North Anna and Surry will not result in a measurable impact on normal operating plant releases and will not increase the predicted radiological consequences of accidents postulated in the Updated Final Safety Analysis Report (UFSAR). Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

The third proposed change, an increase of the Surry Minimum Temperature for Criticality limit from 522 ºF to 538 ºF, would provide Dominion with increased flexibility during loading pattern development as well as improved design margins when coupled with the second proposed change. The Minimum Temperature for Criticality is used within the reload verification process to ensure the assumptions made in the safety analysis remain bounding for the given cycle design. With implementation of the proposed change, the reload design and licensing requirements will remain in place and continue to be met at the increased Minimum Temperature for Criticality limit.

The increase in the Surry Minimum Temperature for Criticality limit will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. Should the reactor coolant system (RCS) temperature fall below the proposed limit, the unit would be in an abnormal condition requiring operator action. The operator actions are not changing as a result of the increased Minimum Temperature for Criticality limit. The increase in the Surry Minimum Temperature for Criticality will not result in a measurable impact on normal operating plant releases and will not increase the predicted radiological consequences of accidents postulated in the UFSAR. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

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

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

The use of the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs and the applicable fuel design limits for DNB ratio (DNBR) does not impact any of the applicable design criteria and the pertinent licensing basis criteria will continue to be met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Setpoint safety analysis evaluations have demonstrated that the use of the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs is acceptable. Design and performance criteria will continue to be met, and no new single failure mechanisms will be created. The use of the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

The increase in the Surry Minimum Temperature for Criticality does not result in any plant design changes. In addition, the minimum temperature at which the reactor is taken critical is not an accident initiator. The nominal average reactor coolant system temperature during an approach to criticality is several degrees higher than the limit proposed for the Minimum Temperature for Criticality.

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

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

The first two proposed changes would allow Dominion to use the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs to perform licensing calculations for North Anna and Surry using the DDLs documented in Appendix D of Fleet Report DOM-NAF-2. North Anna TS 2.1, “Safety Limits,” states that, “the departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5 [COLR].” The DNBR limits meet the design basis of avoiding DNB with 95% probability at a 95% confidence level. Surry TS 2.1, “Safety Limits, Reactor Core,” specifies that “for transients analyzed using the deterministic methodology, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit.” The required DNBR margin of safety for North Anna and Surry, which in this case is the margin between the 95/95 DNBR limit and clad failure, is therefore not reduced. Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

The increased Minimum Temperature for Criticality in conjunction with the appropriate core designs will ensure the current TS limits for the most

> positive moderator temperature coefficient will continue to be satisfied. The current analyses are bounding and remain applicable with the increased Minimum Temperature for Criticality. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

<u>Attorney for licensee</u>: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS-2, Richmond, VA 23219.

<u>NRC Branch Chief</u>: Robert Pascarelli.

Notice of Issuance of Amendments to Facility Operating Licenses and Combined 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.

A notice of consideration of issuance of amendment to facility operating license or combined license, as applicable, 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 NRC's Public Document Room (PDR), located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through the Agencywide Documents Access and Management System (ADAMS) in the NRC Library at 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's Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr.resource@nrc.gov.

Detroit Edison Company, Docket No. 50-16, Enrico Fermi Atomic Power Plant, Unit 1, (Fermi 1) Monroe County, Michigan.

Date of amendment request: December 21, 2012 (ML13002A037).

Brief description of amendment: This amendment revised the Fermi 1 license to change the licensee's name on the license to "DTE Electric Company." This name change is purely administrative in nature. Detroit Edison is a wholly owned subsidiary of DTE Energy Company, and this name change is part of a set of name changes of DTE Energy subsidiaries to conform their names to the “DTE” brand name. No other changes are contained within this amendment. This change does not involve a transfer of control over or of an interest in the license for Fermi 1.

Date of issuance: August 8, 2013.

Effective date: On the date of issuance of this amendment and must be fully implemented no later than 60-calendar days from the date of issuance.

Amendment No.: 21.

Facility Operating License No. DPR-9: Amendment revised the License by replacing “the Detroit Edison” with “DTE Electric” on pages 1, 2, 4, and 5.

Date of initial notice in *Federal Register*: March 19, 2013 (78 FR 16876).

The NRC’s related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 2013.

No significant hazards consideration comments: None received.

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

Date of application for amendments: November 22, 2011, as supplemented by letters dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013.

Brief description of amendments: The amendments revised the Technical Specifications to allow single discharge header operation of the nuclear service water system for a time period of 14 days.

Date of issuance: August 9, 2013.

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

Amendment Nos.: 271 and 267.

Renewed Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the licenses and the Technical Specifications.

Date of initial notice in *Federal Register*: May 15, 2012 (77 FR 28630). The supplements dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013, 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 August 9, 2013

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50-271, Vermont Yankee Nuclear Power Station (VYNPS), Vernon, Vermont

Date of amendment request: December 21, 2012, as supplemented on March 19, April 29, May 7, May 14, and June 26, 2013.

Brief description of amendment: The amendment revised the VYNPS licensing basis relative to how the station satisfies the requirements of 10 CFR 50.63, "Loss of all alternating current power," by replacing the Vernon Hydroelectric Station with an onsite diesel generator as the alternate alternating current power source that would provide acceptable capability to withstand a station blackout under 10 CFR 50.63(c)(2). The change involves revisions to the VYNPS facility and procedures described in the Updated Final Safety Analysis Report.

Date of Issuance: August 15, 2013.

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

Amendment No.: 258.

Facility Operating License No. DPR-28: The amendment revised the License.

Date of initial notice in *Federal Register*: March 19, 2013 (78 FR 16881).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 15, 2013.

No significant hazards consideration comments received: No.

Northern States Power Company - Minnesota (NSPM), Docket No. 50-263, Monticello Nuclear Generating Plant (MNGP), Wright County, Minnesota

Date of application for amendment: December 6, 2012.

Brief description of amendment: The amendment revises the MNGP Technical Specifications (TS) Section 3.10.1. Specifically, the amendment revise Limiting Condition for Operation 3.10.1 and the associated TS Bases to expand its scope to include provisions for temperature excursions greater than 212°F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice or hydrostatic test, while considering operation conditions to be in Mode 4. The changes are consistent with NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-484, Revision 0, "Use of TS 3.10.1 for Scram Time Testing Activities."

Date of issuance: August 9, 2013.

Effective date: This license amendment is effective as of the date of its date of issuance and will be implemented within 120 days of issuance.

Amendment No.: 174.

Renewed Facility Operating License No. DPR-22: Amendment revises the Renewed Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: March 4, 2013.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 2013.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 23rd day of August 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michele G. Evans, Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.