[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2011-0261]

Biweekly Notice

Applications and Amendments to Facility Operating 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 (the Commission or 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 upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 20, 2011 to November 2, 2011. The last biweekly notice was published on November 1, 2011 (76 FR 67485).

ADDRESSES: Please include Docket ID **NRC-2011-0261]** in the subject line of your comments. For additional instructions on submitting comments and instructions on accessing documents related to this action, see “Submitting Comments and Accessing Information” in the **SUPPLEMENTARY INFORMATION** section of this document. You may submit comments by any one of the following methods:

- **Federal Rulemaking Web Site:** Go to *http://www.regulations.gov* and search for documents filed under Docket ID **NRC-2011-0261]**. Address questions about NRC dockets to Carol Gallagher, telephone: 301-492-3668; e-mail: *Carol.Gallagher@nrc.gov*.
- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.
- **Fax comments to:** RADB at 301-492-3446.

SUPPLEMENTARY INFORMATION:

Submitting Comments and Accessing Information

Comments submitted in writing or in electronic form will be posted on the NRC Web site and on the Federal rulemaking Web site, *http://www.regulations.gov*. Because your comments will not be edited to remove any identifying or contact information, the NRC cautions you against including any information in your submission that you do not want to be publicly disclosed.

The NRC requests that any party soliciting or aggregating comments received from other persons for submission to the NRC inform those persons that the NRC will not edit their comments to remove any identifying or contact information, and therefore, they should not include any information in their comments that they do not want publicly disclosed.

You can access publicly available documents related to this document using the following methods:

- **NRC's Public Document Room (PDR):** The public may examine and have copied, for a fee, publicly available documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** Publicly available documents created or received at the NRC are available online in the NRC Library at *http://www.nrc.gov/reading-rm/adams.html*. From this page, the public can gain entry into ADAMS, which provides text and image files of the NRC's public documents. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to *pdr.resource@nrc.gov*.

- **Federal Rulemaking Web Site:** Public comments and supporting materials related to this notice can be found at *http://www.regulations.gov* by searching on Docket ID **NRC-2011-0261]**.

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the NRC's PDR, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. NRC regulations are accessible electronically from the NRC Library on the NRC 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 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 identification

(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 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 guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html*. A filing is considered complete at the time the 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 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 Meta System Help Desk through the "Contact Us" link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail at *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. 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. Non-timely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

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.

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

Date of amendment request: July 5, 2011, as supplemented by letter dated September 12, 2011.

Description of amendment request: The proposed amendment would modify the Millstone Power Station, Unit 3 (MPS3), Technical Specifications (TSs) by relocating specific surveillance frequencies to a licensee-controlled program, the Surveillance Frequency Control Program (SFCP). The proposed changes are based on the Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed TSTF] Initiative 5b" (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML090850642). Plant-specific deviations from TSTF-425 are proposed to accommodate differences between the MPS3 TSs and the model TSs originally used to develop TSTF-425. The proposed plant-specific deviations involve fixed periodic frequency surveillances, and are therefore consistent with TSTF-425, and editorial deviations.

The NRC staff issued a Notice of Availability for TSTF-425 in the *Federal Register* on July 6, 2009 (74 FR 31996). The notice included a model safety evaluation and a model no significant hazards consideration (NSHC) determination. In its application dated July 5, 2011, as supplemented by letter dated September 12, 2011, Dominion Nuclear Connecticut, Inc. (DNC or the licensee) provided its analysis of the issue of NSHC based on the model NSHC determination for TSTF-425.

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

   Response: No.

   The proposed changes relocate the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the TSs for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

   Response: No.

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

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

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

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Dominion will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI [Nuclear Energy Institute] 04-10, Rev. 1, ["Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies,"] in accordance with the TS SFCP [Surveillance Frequency Control Program]. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177 ["An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications"].

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

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

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

NRC Branch Chief: Harold K. Chernoff.

Entergy Nuclear Operations, Inc., Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: July 22, 2011.

<u>Description of amendment request</u>: The proposed amendment would modify the Technical Specifications (TS) by relocating specific Surveillance Frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)-425, Revision 3, ‘‘Relocate Surveillance Frequencies to Licensee Control-Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b.’’

The existing Bases information describing the basis for the Surveillance Frequency will be relocated to the licensee-controlled Surveillance Frequency Control Program. Additionally, the change would add a new program, TS 5.5.15, “Surveillance Frequency Control Program,” to TS Section 5.5, “Programs and Manuals.”

The changes are consistent with NRC approved TSTF-425, Revision 3, (Rev. 3) (ADAMS Package Accession No. ML090850642). The *Federal Register* notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

<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 change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   The proposed change relocates the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Entergy will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the TS SFCP [Surveillance Frequency Control Program]. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

   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.

<u>Attorney for licensee</u>: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

<u>NRC Branch Chief</u>: Nancy L. Salgado.

Entergy Nuclear Operations, Inc., Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Renewed Facility Operating License No. DPR-059

Date of amendment request: August 16, 2011.

Description of amendment request: The proposed amendment to the Renewed Facility Operating License would revise the James A. FitzPatrick Nuclear Power Plant (JAF) current licensing basis (CLB) to allow the use of On Load Tap Changers (OLTCs) with new Reserve Station Service Transformers (RSST) that provide offsite power to JAF.

The OLTCs are sub-components of two new RSSTs that will be installed at JAF in September 2012, during the scheduled refueling outage. The OLTCs are designed to compensate for offsite voltage variations and will provide added assurance that acceptable bus voltage is maintained for safety-related equipment.

The proposed amendment requests NRC approval to operate the OLTCs in the automatic mode. Operation of the OLTCs in the automatic mode was evaluated under 10 CFR 50.59 and it was determined that it requires NRC approval because such operation creates the possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The proposed amendment would change the UFSAR and the Technical Specification (TS) Bases. There would be no changes to the plant TS associated with this request.

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 operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment will allow operation of the OLTCs in automatic mode. The only accident previously evaluated where the probability of an accident is potentially affected by the change is the loss of offsite power (LOOP) Abnormal Operational Transient (AOT). Failure of an OLTCs while in the automatic mode of operation that results in decreased voltage to the engineered safety features (ESF) buses could cause a LOOP if voltage decreased below the degraded voltage relay (DVR) setpoint. The two postulated failure scenarios are: 1) failure of an [a] primary microcontroller that results in rapidly decreasing voltage supplied to the ESF buses and; 2) failure of an [a] primary microcontroller to respond to decreasing grid voltage. For the first scenario, a backup microcontroller is provided for each OLTC, which makes this failure unlikely. For the second scenario, since grid voltage changes typically occur relatively slowly and the magnitude of the resulting change would be limited to the effect of the change in grid voltage, operators would have ample time to address the condition utilizing identified procedures. In addition, the frequency of occurrence of these failure modes is small, based on the operating history of similar equipment at other plants. Furthermore, in both of the above potential failure modes, operators can take manual control of the OLTC to mitigate the effects of the failure. Thus, the probability of a LOOP will not be significantly increased by operation of the OLTCs in the automatic mode.

The proposed amendment has no effect on the consequences of a LOOP, since the emergency diesel generators (EDGs) provide power to safety-related equipment following a LOOP. The design and function of the EDGs are not affected by the proposed change. The probability of other previously evaluated accidents is not affected, since the proposed amendment does not affect the way plant equipment is operated and thus does not contribute to the initiation of any of the previously evaluated accidents. The OLTC is equipped with a backup microcontroller, which inhibits gross improper action of the OLTC in the event of primary microcontroller failure. Additionally, the operator has procedurally identified actions available to prevent a sustained high voltage condition from occurring. Damage due to overvoltage is time-dependent, requiring a sustained high voltage condition. Therefore, damage to safety-related equipment is unlikely, and the consequences of previously evaluated accidents are not significantly increased. Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment involves electrical transformers that provide offsite power to safety-related equipment for accident mitigation. The proposed change does not alter the design, physical configuration, or mode of operation of any other plant structure, system, or component. No physical changes are being made to any other portion of the plant, so no new accident causal mechanisms are being introduced. Although the proposed change potentially affects the consequences of previously evaluated accidents (as discussed in the response to Question 1), it does not result in any new mechanisms that could initiate damage to the reactor or its principal safety barriers (i.e., fuel cladding, reactor coolant system, or primary containment).

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment does not affect the inputs or assumptions of any of the analyses that demonstrate the integrity of the fuel cladding, reactor coolant system, or containment during accident conditions. The allowable values for the degraded voltage protection function are unchanged and will continue to ensure that the degraded voltage protection function actuates when required, but does not actuate prematurely to unnecessarily transfer safety-related loads from offsite power to the emergency diesel generators. Automatic operation of the OLTCs increases the margin of safety by reducing the potential for transferring loads to the EDGs during an under voltage or over voltage event on the offsite power sources.

Therefore, the proposed amendment to the JAF design basis 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. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Nancy L. Salgado.

Entergy Nuclear Operations, Inc., Docket No. 50-255, Palisades Nuclear Plant (PNP), Van Buren County, Michigan

Date of amendment request: August 16, 2011, as supplemented by letter dated October 6, 2011.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 5.5.14, “Containment Leak Rate Testing Program” to increase the value of the calculated peak containment internal pressure from 53 pounds per square inch gauge (psig) to 54.2 psig. This increase is due to an increase in the calculated mass and energy release during the blowdown phase of the design basis loss-of-coolant accident (LOCA). The increase in the predicted mass and energy release is due to the correction of an error in the calculation of the current value of P_a. The regulations at 10 CFR Part 50 Appendix J Option B define P_a as the calculated peak containment internal pressure related to the design basis LOCA as specified in the TS and specifies the requirements for containment leakage rate testing.

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

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

   Response: No.

   The proposed change to P_a does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change.

   The change in P_a will not affect radiological dose consequence analyses. PNP radiological dose consequence analyses assume a certain containment atmosphere leak rate based on the maximum allowable containment leakage rate, which is not affected by the change in calculated peak containment internal pressure. The Appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The consequences of an accident previously evaluated will not be increased by this proposed change.

Therefore, operation of the facility in accordance with the proposed change to P_a will 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 provides a higher P_a than currently described in the TS. This change is a result of an increase in the mass and energy release input for the loss of coolant accident containment response analysis. The calculated peak containment pressure remains below the containment design pressure of 55 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Therefore, operation of the facility in accordance with the proposed change to TS Section 5.5.14 would not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The calculated peak containment pressure remains below the containment design pressure of 55 psig. Since PNP radiological consequence analyses are based on the maximum allowable containment leakage rate, which is not being revised, the change in the calculated peak containment pressure does not represent a significant change in the margin of safety.

   Therefore, operation of the facility in accordance with the proposed change to TS Section 5.5.14 does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. William Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Ave., White Plains, NY 10601.

NRC Branch Chief: Robert J. Pascarelli.

NextEra Energy Duane Arnold, LLC, Docket No. 50-331, Duane Arnold Energy Center (DAEC),Linn County, Iowa

Date of amendment request: May 31, 2011.

Description of amendment request: The proposed amendment would upgrade selected DAEC Emergency Action Levels (EALs) based on NEI 99-01, Revision 5, "Methodology for Development of Emergency Action Levels," using the guidance of NRC Regulatory Issue Summary 2003-18, Supplement 2, "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels." NextEra Energy Duane Arnold currently uses an emergency classification scheme based on NEI 99-01, Revision 4.

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 any accident previously evaluated?

   Response: No.

   These changes affect the DAEC Emergency Plan and do not alter any of the requirements of the Operating License or the Technical Specifications. The proposed changes do not modify any plant equipment and do not impact any failure modes that could lead to an accident. Additionally, the proposed changes do not impact the consequence of any analyzed accident since the changes do not affect any equipment related to accident mitigation.

   Based on this discussion, the proposed amendment does not increase 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 previously evaluated?

   Response: No.

   These changes affect the DAEC Emergency Plan and do not alter any of the requirements of the Operating License or the Technical Specifications. They do not modify any plant equipment and there is no impact on the capability of the existing equipment to perform their intended functions. No system setpoints are being modified and no changes are being made to the method in which plant operations are conducted. No new failure modes are introduced by the proposed changes. The proposed amendment does not introduce accident initiator or malfunctions that would cause a new or different kind of accident.

   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 the margin of safety?

   Response: No.

   These changes affect the DAEC Emergency Plan and do not alter any of the requirements of the Operating License or the Technical Specifications. The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety.

   Therefore, the proposed changes will not result in a significant reduction in the margin of safety as defined in the bases for technical specifications covered in this license amendment request.

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. Marjan Mashhadi, 801 Pennsylvania Avenue, NW, Suite 220. Washington, DC 20004.

NRC Branch Chief: Robert J. Pascarelli.

South Carolina Electric and Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station (VCSNS), Unit 1, Fairfield County, South Carolina

Date of amendment request: August 23, 2011.

Description of amendment request: The proposed amendment would delete the license condition, 2.G.1 of the Facility Operating License, that requires reporting of violations of Section 2.C of the Facility Operating License consistent with the *Federal Register* notice dated November 4, 2005 (70 FR 67202) as part of the consolidated line item improvement process (CLIIP). The proposed amendment would also delete a reporting requirement in the VCSNS Technical Specifications (TS), Section 6.6, which is duplicative of NRC regulations, and make appropriate adjustments to the TS index to reflect that deletion.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has referenced the NRC staffs model no significant hazards consideration, presented in a *Federal Register* notice (70 FR 51098; August 29, 2005), and made available for use by *Federal Register* notice (70 FR 67202; November 4, 2005), and is presented below:

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

   Response: No.

   The proposed change involves the deletion of a reporting requirement. The change does not affect plant equipment or operating practices and therefore does not significantly increase 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 proposed change is administrative in that it deletes a reporting requirement. The change does not add new plant equipment, change existing plant equipment, or affect the operating practices of the facility. Therefore, the 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 deletes a reporting requirement. The change does not affect plant equipment or operating practices and therefore does not involve a significant reduction in a margin of safety.

Based on the above, the NRC staff proposes that the change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

Attorney for licensee: J. Hagood Hamilton, Jr., South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Branch Chief: Gloria Kulesa.

Tennessee Valley Authority (TVA), Docket No. 50-328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of amendment request: August 31, 2011 (TS-SQN-2011-03).

Description of amendment request: During Sequoyah Nuclear Plant (SQN), Unit 2, spring 2011 refueling outage (RFO), two penetrations through the shield building (SB) dome were created. To maintain SB integrity, these penetrations were closed with a steel hatch assembly prior to entering Mode 4 at the end of the RFO. The proposed amendment would temporarily revise the technical specifications to allow opening of one of the penetration hatches in the SB dome for up to 5 hours per day, 6 days per calendar week while in Modes 1 through 4 during SQN, Unit 2 Cycle 18, and until entering Mode 5 at the start of the SQN, Unit 2 fall 2012 RFO. The two approximately 18-inch diameter penetrations on the SB dome will provide steam generator replacement project workers an alternate path of moving materials inside the annulus for online work. Without use of the SB dome penetration hatches, materials would travel through the auxiliary building (AB), to the annulus access door, and be hoisted up the annual access ladders. Bypassing the AB and the annulus access ladders reduces the risk of potential adverse effects to sensitive equipment along the path. The alternate path is estimated to save approximately 2.8 roentgen equivalent man by allowing materials to be passed through the open SB dome penetration hatch in lieu of carrying the material past higher dose areas. In addition, passing material through the open SB dome hatch will significantly improve the industrial safety aspect of the work and will provide work efficiency gains since material will be provided closer to the point of use.

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 bounding transients and accidents (i.e., loss-of-coolant-accident (LOCA), tornado, and earthquake) that are potentially affected by the assumptions associated with the use of one of the Shield Building dome penetration hatches (2-EQH-410-0010 or 2-EQH-410-0011) have been evaluated/analyzed. Weather and seismic related events are determined by regional conditions. Therefore, the probability of a tornado or earthquake is not affected by the use of one of the Shield Building dome penetration hatches. Failure of the Shield Building or Emergency Gas Treatment System (EGTS) is not an initiator of any of the accidents and transients described in the Updated Final Safety Analysis Report (UFSAR). Therefore, since no initiating event mechanisms are being changed, the use of one of the Shield Building dome penetration hatches will not result in an increase in probability of any previously evaluated accident.

   The use of one of the Shield Building dome penetration hatches affects the integrity of the Shield Building and the ability of the EGTS to maintain the annulus at a negative pressure relative to the outside atmosphere such that the function in mitigating the radiological consequences of an accident is affected. TVA's evaluation documents the radiological consequences of a LOCA assuming the open Shield Building dome penetration hatch is closed within 22.1 minutes and the operating EGTS trains draw down the annulus to -0.25 inches wg [water gauge] to effectively end the direct release of radionuclides to the environment 23.1 minutes after accident initiation. TVA's evaluation also documents the mission dose an individual may receive during ingress from the Control Building Habitability area to the Shield Building dome, closure of the steel hatch assembly, and egress from the Shield Building dome. Although the LOCA radiological consequences with the Shield Building dome penetration hatch open for 22.1 minutes (and assumed to be a direct release path for 23.1 minutes) are higher than those described in the UFSAR, the offsite and Control Room doses remain within the limits of 10 CFR 50.67, "Accident source term," when applying the Alternate Source Term (AST) methodology in accordance with Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000. The calculated mission doses are also less than the limits of 10 CFR 50.67, "Accident source term," paragraph (b)(2)(iii) when applying the AST methodology in accordance with Regulatory Guide 1.183.

Therefore, since the increase in radiological consequences of the previously evaluated LOCA remains bounded by the applicable regulatory limits, the increased consequences are not considered significant.

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

   Response: No.

   Loss of Shield Building integrity or EGTS failure is not an initiator of any of the accidents and transients described in the UFSAR. Shield Building integrity as the pressure boundary for the EGTS, and loss of Shield Building integrity due to an open penetration hatch in the Shield Building dome (Hatch 2-EQH-410-0010 or 2-EQH-410-0011) during Modes 1 through 4 potentially renders both trains of EGTS incapable of establishing a post-accident annulus pressure. This condition would require SQN, Unit 2, to enter the Action of TS [Technical Specification] Limiting Condition for Operation (LCO) 3.6.1.8 (for the condition of one train of EGTS being inoperable) and enter TS LCO 3.0.3 (due to both trains of EGTS being inoperable). TS LCO 3.0.3 requires that the unit be shutdown within specified time periods. Closure of the open Shield Building dome penetration steel hatch assembly restores the integrity of the Shield Building such that both trains of EGTS would be operable as required by TS LCO 3.6.1.8. Failure of the Shield Building dome penetration steel hatch assemblies will not initiate any of the accidents and transients described in the UFSAR. Postulated failures of the Shield Building dome penetration steel hatch assemblies are degradation/damage to the seals or damage to the hatch hinges. Like any other Shield Building failure during Modes 1 through 4 that potentially renders both trains of EGTS inoperable, these postulated Shield Building dome penetration steel hatch assembly failures result in a loss of Shield Building integrity and require that the failed component be repaired or replaced within a specified time period or that plant shutdown be initiated.

   Therefore, a failure of a steel hatch assembly during use of the Shield Building dome penetration will not initiate an accident nor create any new failure mechanisms. The changes do not result in any event previously deemed incredible being made credible. The use of Shield Building dome Penetration Hatch 2-EQH-410-0010 or 2-EQH-410-0011 is not expected to result in more adverse conditions in the annulus and is not expected to result in any increase in the challenges to safety systems.

Manual action is required to close an open Shield Building dome penetration hatch and to configure the EGTS control loops following the opening and closing of a Shield Building dome penetration hatch such that the EGTS will respond as designed. NRC Information Notice (IN) 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," and American National Standards Institute/American Nuclear Society (ANSI/ANS)-58.8, "Time Response Design Criteria for Safety-Related Operator Actions," provide guidance for consideration of safety-related operator actions.

The manual actions implemented as a result of this change can be completed within the guidance and criteria provided in Information Notice (IN) 97-78 and ANSI/ANS-58.8. Consequently, the manual actions can be credited in the mitigation of events that require Shield Building integrity. With credit for the manual actions to close an open Shield Building dome penetration hatch (2-EQH-410-0010 or 2-EQH-410-0011) and reconfigure the EGTS control loops subsequent to an event, the types of accidents currently evaluated in the UFSAR remain the same.

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

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

   Response: No.

   The manual actions to close an open Shield Building dome penetration hatch (2-EQH-410-0010 or 2-EQH-410-0011) and to configure the EGTS control loops following the opening and closing of a Shield Building dome penetration hatch ensure that the EGTS will respond as designed. Safety-related instrumentation is available to inform operators that a reactor trip has occurred, and dedicated trained individuals will be positioned to close an open Shield Building dome penetration hatch should an accident occur. The manual actions meet the criteria for safety-related operator actions contained in NRC IN 97-78 and ANSI/ANS-58.8. The use of manual actions maintains the margin of safety by assuring compliance with acceptance limits reviewed and approved by the NRC. The appropriate acceptance criteria for the various analyses and evaluations have been met; therefore, there has not been a reduction in any margin of safety.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, 6A West Tower, Knoxville, Tennessee 37902.

NRC Branch Chief: Douglas A. Broaddus.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the 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 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 Reference staff at 1-800-397-4209, 301-415-4737 or by email to *pdr.resource@nrc.gov*.

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

Date of application for amendment: April 29, 2011.

Brief description of amendment: The amendments will modify Technical Specification (TS) to define a new time limit for restoring inoperable reactor coolant system (RCS) leakage detection instrumentation to operable status and establish alternative methods of monitoring RCS leakage when one or more require monitors are inoperable. The changes are consistent with Nuclear Regulatory Commission-approved Technical Specification Task Force Traveler-513, Revision 3. The availability of this TS improvement was published in the *Federal Register* on January 3, 2011 (76 FR 189), as part of the consolidated line item improvement process.

Date of issuance: October 25, 2011.

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

Amendment Nos: 288 and 175

Renewed Facility Operating License Nos. DPR-66 and NPF-73: The amendments revised the License and TS.

Date of initial notice in *Federal Register*: July 12, 2011 (76 FR 40940).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2011.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 5, 2010, supplemented by letters dated February 22, May 20, September 14, and September 22, 2011.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.5.1 Fuel Storage - Criticality, to include new spent fuel storage patterns that account for both the increase in fuel maximum enrichment from 4.5 weight (wt) percent (%) U-235 to 5.0 wt% U-235 and the impact on the fuel of higher power operation proposed under the Extended Power Uprate license amendment request. Although the fuel storage has been analyzed at the higher fuel enrichment in the new criticality analysis, the fuel enrichment limit of 4.5 wt% U-235 specified in TS 5.5.1 will not be changed with the issuance of these license amendments.

Date of issuance: October 31, 2011.

Effective date: As of the date of issuance and shall be implemented by the completion of the Cycle 26 refueling outage for Unit 3 and Cycle 27 refueling outage for Unit 4.

Amendment Nos.: Unit 3 - 246 and Unit 4 - 242.

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

Date of initial notice in *Federal Register*: October 5, 2010 (75 FR 61527). The supplements dated February 22, May 20, September 14, and September 22, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2011.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 29, 2010, as supplemented by letters dated June 10 and August 31, 2011.

Brief description of amendment: The amendment revised the acceptance criteria in CNS Technical Specification (TS) 3.8.4, “DC [Direct Current] Sources - Operating,” Surveillance Requirement (SR) 3.8.4.1, and TS 3.8.6, “Battery Cell Parameters,” Table 3.8.6-1, “Battery Cell Parameter Requirements.” Specifically, amendment revised the acceptance criteria in TS SR 3.8.4.1 and TS Table 3.8.6-1 by revising the battery terminal voltage on float charge and specific gravity acceptance criteria to ensure that the safety-related batteries can perform their safety functions and will remain operable during postulated design basis events.

Date of issuance: October 28, 2011.

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

Amendment No.: 239.

Renewed Facility Operating License No. DPR-46: Amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: January 25, 2011 (76 FR 4386). The supplemental letters dated June 10 and August 31, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 28, 2011.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 2, 2010, as supplemented on January 27, 2011.

Brief description of amendment: The amendment revises the NMP1 Technical Specification (TS) Section 3.6.2, "Protective Instrumentation," by modifying the operability requirements for the average power range monitoring (APRM) instrumentation system. The amendment eliminates the requirements that the APRM "Upscale" and "Inoperative" scram and control rod withdrawal block functions be operable when the reactor mode switch is in the Refuel position. The amendment also clarifies the operability requirements for the APRM "Downscale" control rod withdrawal block function when the reactor mode switch is in the Startup and Refuel positions.

Date of issuance: October 31, 2011.

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

Amendment No.: 211.

Renewed Facility Operating License No. DPR-63: The amendment revises the License and TSs.

Date of initial notice in *Federal Register*: March 22, 2011 (76 FR 16007). The supplemental letter dated January 27, 2011, provided additional information that clarified the application and did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2011.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 30, 2010, as supplemented on June 1 and December 29, 2010, and January 14, February 25, April 27, and July 25, 2011.

Brief description of amendment: The amendment changes the NMP2 Technical Specification (TS) 3.8.1, "AC Sources - Operating," to extend the Completion Time (CT) for an inoperable Division 1 or Division 2 diesel generator (DG) from 72 hours to 14 days.

Date of issuance: October 31, 2011.

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

Amendment No.: 138.

Renewed Facility Operating License No. NPF-069: The amendment revises the License and TSs.

Date of initial notice in *Federal Register*: July 13, 2010 (75 FR 39980). The supplemental letters dated June 1 and December 29, 2010, and January 14, February 25, April 27, and July 25, 2011, provided additional information that clarified the application and did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2011.

No significant hazards consideration comments received: No.

Northern States Power Company - Minnesota, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: December 22, 2009, as supplemented by letters dated July 23, 2010, August 20, 2010, October 8, 2010, January 14, 2011, February 23, 2011, April 6, 2011, and August 9, 2011.

Brief description of amendments: The amendments approve the application of the leak-before-break methodology to certain piping systems attached to the reactor coolant system at the Prairie Island Nuclear Generating Plant, Units 1 and 2.

Date of issuance: October 27, 2011.

Effective date: As of the date of issuance. The amendment for Unit 1 shall be implemented within 180 days. The amendment for Unit 2 shall be implemented before the end of the next scheduled Unit 2 refueling outage.

Amendment Nos.: 204, 191.

Facility Operating License Nos. DPR-42 and DPR-60: Amendments revised the Renewed Facility Operating Licenses.

Date of initial notice in *Federal Register*: May 11, 2010 (75 FR 26290). The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 27, 2011.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 7th day of November 2011.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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