proposals as part of the selection process for awards. The review and evaluation may also include assessment of the progress of awarded proposals. The majority of these meetings will take place at NSF, 4201 Wilson, Blvd., Arlington, Virginia 22230.

These meetings will be closed to the public. The proposals being reviewed include information of a proprietary or confidential nature, including technical information; financial data, such as salaries; and personal information concerning individuals associated with the proposals. These matters are exempt under 5 U.S.C. 552b(c), (4) and (6) of the Government in the Sunshine Act. NSF will continue to review the agenda and merits of each meeting for overall compliance of the Federal Advisory Committee Act.

These closed proposal review meetings will not be announced on an individual basis in the **Federal Register**. NSF intends to publish a notice similar to this on a quarterly basis. For an advance listing of the closed proposal review meetings that include the names of the proposal review panel and the time, date, place, and any information on changes, corrections, or cancellations, please visit the NSF Web site: *http://www.nsf.gov/events/.* This information may also be requested by telephoning Crystal Robinson at 703/292–8687.

Dated: July 31, 2014.

Suzanne Plimpton,

Acting Committee Management Officer.

[FR Doc. 2014–18448 Filed 8–4–14; 8:45 am]

BILLING CODE 7555–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2014–0180]

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

SUMMARY: 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 to 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 July 10, 2014 to July 23, 2014.

DATES: Comments must be filed by September 4, 2014. A request for a hearing must be filed by October 6, 2014.

ADDRESSES: You may submit comments 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–2014–0180. Address questions about NRC dockets to Carol Gallagher; telephone: 301–287–3422; email: *Carol.Gallagher@nrc.gov.*
- Mail comments to: Cindy Bladey, Office of Administration, Mail Stop: 3WFN–06–A44M, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

For additional direction on obtaining information and submitting comments, see ''Obtaining Information and Submitting Comments'' in the **SUPPLEMENTARY INFORMATION** section of this document.

FOR FURTHER INFORMATION CONTACT: Shirley Rohrer, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555–0001; telephone: 301–415–5411, email: *Shirley.Rohrer@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC–2014–0180 when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- Federal rulemaking Web site: Go to *http://www.regulations.gov* and search for Docket ID NRC–2014–0180.
- NRC's Agencywide Documents Access and Management System (ADAMS): You may obtain publicly-available documents online in the ADAMS Public Documents collection 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 email to *pdr.resource@nrc.gov.* The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in the **SUPPLEMENTARY INFORMATION** section
- 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–2014–0180 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 in comment submissions that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at *http://www.regulations.gov* as well as enter the comment submissions into ADAMS, and 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 into ADAMS.

II. Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined Licenses and Proposed No Significant Hazards Consideration Determination

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.

A. Opportunity To Request a Hearing and Petition for Leave To Intervene

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's 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.

B. Electronic Submissions (E-Filing)

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 ten 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by email 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/getting-started.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 NRC 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 email notice confirming receipt of the document. The E-Filing system also distributes an email 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 NRC'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's public Web site at *http://www.nrc.gov/site-help/e-submittals.html,* by email 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 these license amendment applications, see the application for amendment which is available for public inspection in ADAMS and at the NRC's PDR. For additional direction on accessing information related to this document, see the ''Obtaining Information and Submitting Comments'' section of this document.

Dominion Energy Kewaunee (DEK), Docket No. 50–305, Kewaunee Power Station (KPS), Kewaunee County, Wisconsin

Date of amendment request: January 16, 2014. A publicly-available version is in ADAMS under Accession No. ML14029A076.

Description of amendment request: The proposed amendment would modify the KPS renewed facility operating license by revising the emergency plan and the associated emergency action level (EAL) scheme consistent with the KPS permanent shutdown and defueled status. On February 25, 2013, DEK submitted a certification of permanent cessation of power operations pursuant to 10 CFR, Part 50, Section 50.82(a)(1)(i), stating that DEK had decided to permanently cease power operation of KPS on May 7, 2013. With the docketing of subsequent certification for permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii) on May 14, 2013, the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2). The proposed changes to the emergency plan and EAL scheme are being submitted to the U.S. Nuclear Regulatory Commission (NRC) for approval prior to implementation, as required under 10 CFR 50.54(q)(4) and 10 CFR Part 50, Appendix E, Section IV.B.2.

DEK states that the proposed emergency plan changes do not meet all the standards of 10 CFR 50.47(b) and requirements of 10 CFR Part 50, Appendix E. By letter dated July 31, 2013 (ADAMS Accession No. ML13221A182), DEK submitted requests to the NRC for exemptions from portions of 10 CFR 50.47(b), 10 CFR 50.47(c)(2), and 10 CFR Part 50, Appendix E, Section IV, that the proposed emergency plan does not meet. The proposed emergency plan revision is predicated on the approval of the requested exemptions.

Basis for proposed no significant hazards consideration determination: Pursuant to 10 CFR 50.92, the NRC staff 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.

KPS has permanently ceased operation and is permanently defueled. Because the 10 CFR Part 50 license for KPS no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel, as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. Analyses of the remaining credible accidents, as documented in the KPS Updated Safety Analysis Report (USAR), show that any releases beyond the site boundary would be below the Environmental Protection Agency (EPA) Protective Action Guides (PAGs) exposure levels, as detailed in the EPA's ''Protective Action Guide and Planning Guidance for Radiological Incidents,'' Draft for Interim Use and Public Comment dated March 2013.

The proposed amendment would revise the emergency plan and EAL scheme to reflect the permanently defueled status of the plant. The proposed changes discontinue offsite emergency planning requirements and reduce the scope of onsite emergency planning requirements by removing positions that are no longer credited or needed for the remaining credible design basis accidents. The revised emergency plan and EAL scheme focus on responding to the emergencies that may arise from off-normal events and conditions which could indicate a degradation of the level of safety or indicate a security threat bounded by the type and significance of the remaining credible design basis accidents in a permanently shutdown and defueled condition.

The proposed changes to the emergency plan do not impact the function of plant structures, systems, or components (SSCs). The proposed changes do not affect accident initiators or precursors, nor do they alter design assumptions. Therefore, the proposed changes to the emergency plan do not involve an increase in the probability of an accident previously evaluated.

The proposed changes to the emergency plan remove positions from the emergency plan that are no longer credited or needed for the remaining credible design basis accidents. The proposed changes do not prevent the ability of the emergency response organization to perform its intended functions to mitigate the onsite consequences of an event for the remaining credible design basis accidents. The proposed changes do not increase the types or amounts of effluent releases beyond the site boundary from the remaining credible design basis accidents.

Therefore, the proposed changes to the emergency plan do not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes to the EAL scheme limit the emergency classification levels to an Unusual Event and Alert. Because no remaining credible accidents can result in releases beyond the site boundary that exceed EPA PAG exposure levels, the need for emergency classifications of Site Area Emergency or General Emergency would not be required at a permanently shutdown and defueled facility. The changes to the EAL scheme do not involve any physical plant changes. The EALs and installed EAL equipment are not accident initiators and therefore the proposed changes to the EAL scheme do not involve an increase in the probability of an accident previously evaluated.

The proposed EAL scheme changes do not affect the capability of SSCs to mitigate a design basis accident. Thus, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed amendment would revise the emergency plan and EAL scheme to reflect the permanently defueled status of the plant. The proposed changes do not involve installation of new equipment or modification of existing equipment, so that no new equipment failure modes are introduced. Also, the proposed changes do not result in a change to the way that the equipment or facility is operated so that no new accident initiators are created.

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

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

Response: No.

The proposed amendment would revise the emergency plan and EAL scheme to reflect the permanently defueled status of the plant. The proposed changes to the emergency plan and EAL scheme do not involve a change in the plant's design, configuration, or operation. The proposed changes do not affect the way the plant structures, systems, and components perform their safety functions or their design margins as they apply to the remaining credible accidents. The proposed changes do not involve a change to the technical specifications. Because there is no change to the physical design or operation of the plant, no change to the accident analyses, and no change to the safety analysis acceptance criteria as a result of this amendment, there is no change to any of these margins.

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

Based on the NRC staff's analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Douglas A. Broaddus.

Duke Energy Carolinas, LLC, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station (ONS), Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: May 20, 2014. A publicly available version is in ADAMS under Accession No. ML14141A415.

Description of amendment request: The proposed amendment requests removal of Technical Specification requirements for ONS units that did not have the Reactor Protection System (RPS)/Engineered Safeguards Protective System (ESPS) digital upgrades or Low Pressure Service Water (LPSW) Reactor Building (RB) Waterhammer Prevention System (WPS) modifications. The Licensee stated that these Technical Specification requirements no longer pertain to ONS since the RPS/ESPS digital upgrade and the LPSW RB WPS modification have been implemented for all three ONS units. The proposed amendment also deletes a Note statement for the Emergency Condenser Circulating Water (ECCW) System Technical Specification that states the Technical Specification is not applicable until after completion of the Service Water upgrade modifications on each respective ONS unit. The licensee stated that the Service Water upgrade modifications have been implemented for each ONS unit.

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

1. Does the Proposed Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

Response: No.

The proposed changes to Technical Specifications 3.3.1, 3.3.3, 3.3.5, 3.3.7, 3.3.27, 3.6.5, 3.7.7, and 3.7.8 do not modify the Reactor Protective System (RPS), Engineered Safeguards Protective System (ESPS), Low Pressure Service Water (LPSW) System, the LPSW Reactor Building (RB) Waterhammer Protection System (WPS) or the Emergency Condenser Circulating Water (ECCW) System, nor make any physical changes to the facility design, material, or construction standards. The proposed changes remove obsolete information from the Technical Specifications that no longer apply to ONS; delete Surveillance Requirements (SRs) for the RPS RB High Pressure trip function and the ESPS RB Pressure—High High actuation parameter that are not applicable; and correct a wording error in a Condition statement for TS 3.7.7 which results in a more stringent Condition. Since the removed information no longer applies to ONS, and the deleted SRs are for equipment features that do not exist for the RPS RB High Pressure trip function and the ESPS RB Pressure—High High actuation parameter, removal of the information and deletion of the SRs do not result in operation that will increase the probability of initiating an analyzed event. Likewise, the more restrictive requirement in the corrected Condition statement continues to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The proposed Technical Specification changes do not alter assumptions relative to mitigation of an accident or transient event. The removal of the obsolete Technical Specification information, deletion of SRs for features that do not exist, and correction of the Technical Specification Condition statement have no effect on the process variables, structures, systems, and components that must be maintained consistent with the safety analyses and licensing basis. Therefore, the proposed Technical Specification 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 changes to Technical Specifications 3.3.1, 3.3.3, 3.3.5, 3.3.7, 3.3.27, 3.6.5, 3.7.7, and 3.7.8 only remove obsolete information from the Technical Specifications pertaining to the RPS/ESPS digital upgrade, the LPSW RB WPS modification installation, and the ECCW System Service Water upgrade modification completion. The proposed changes also delete SRs that verify features that do not exist for the RPS RB High Pressure trip function and the ESPS RB Pressure—High High actuation parameter. Lastly, the proposed changes correct a wording error in a Condition statement for TS 3.7.7 which results in a more stringent Condition. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The RPS, ESPS, LPSW System, LPSW RB WPS, and ECCW System are not associated with any design accident initiation; they only mitigate accidents. However, these proposed Technical Specification changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

3. Does the Proposed Change Involve a Significant Reduction in a Margin of Safety?

Response: No.

The proposed changes to Technical Specifications 3.3.1, 3.3.3, 3.3.5, 3.3.7, 3.3.27, 3.6.5, 3.7.7, and 3.7.8 remove information from the Technical Specifications pertaining to the RPS/ESPS digital upgrade, the LPSW RB WPS modification installation, and the ECCW System Service Water upgrade modification completion. The proposed changes also delete SRs that verify features that do not exist for the RPS RB High Pressure trip function and the ESPS RB Pressure—High High actuation parameter. Lastly, the proposed changes correct a wording error in a Condition statement for TS 3.7.7 which results in a more stringent Condition. The removed Technical Specification information no longer applies to ONS operation and is considered obsolete; the deleted SRs cannot be performed since the affected plant equipment will not support SR testing by design; and the corrected TS 3.7.7 Condition statement results in a more conservative Technical Specification. Removal of the Technical Specification obsolete information has no impact on the margin of safety since the equipment that the Technical Specification information applied to no longer exists at ONS. Deletion of SRs on the subject RPS/ESPS equipment has no impact on the margin of safety since the RPS/ESPS equipment, by design, will not support SR testing. Correction of the TS 3.7.7 Condition statement has no impact on the margin of safety since the correction results in a more conservative Technical Specification. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the proposed Technical Specification 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: Lara S. Nichols, Deputy General Counsel, Duke Energy Corporation, 526 South Church Street—EC07H, Charlotte, NC 28202–1802.

NRC Branch Chief: Robert J. Pascarelli.

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

Date of amendment request: November 11, 2013. A publicly-available version is in ADAMS under Accession No. ML13316C052.

Description of amendment request: Entergy Operations, Inc. (the licensee), has proposed to change the Waterford Steam Electric Station, Unit 3 Updated Final Safety Analysis Report (UFSAR). This change will clarify in the UFSAR how the pressurizer heaters function is met for natural circulation at the onset of a loss-of-offsite power concurrent with the specific single point vulnerability.

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 would describe the specific common circuit breaker associated with the control power closing circuitry to the Switchgears 32A and 32B Supply Circuit Breakers in UFSAR 1.9.26 and 5.4.10 as contained in Attachment 2 [of the licensee's letter dated November 11, 2013] and that local manual operation outside of the Control Room would be necessary to reenergize Pressurizer Heaters during a loss of offsite power concurrent with the specific common circuit breaker being open. Plant Operators are trained and have procedural guidance including manual operator action to address Natural Circulation Cooldown with a Loss of Offsite Power. The Pressurizer Heaters are not themselves a credible initiator of any accident, and the requested amendment makes no change to the Pressurizer Heaters themselves, so the probability of an accident will not be increased. The proposed change would not change the source term nor adversely impact any mitigating systems, so the consequences of an accident will not be increased.

Therefore, the probability or consequences of any accident previously evaluated will not be increased by the proposed change.

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 would describe the specific common circuit breaker associated with the control power closing circuitry to the Switchgears 32A and 32B Supply Circuit Breakers in UFSAR 1.9.26 and 5.4.10 as contained in Attachment 2 [of the licensee's letter dated November 11, 2013] and that local manual operation outside of the Control Room would be necessary to reenergize Pressurizer Heaters during a loss of offsite power concurrent with the specific common circuit breaker being open.

The proposed changes do not involve a change in the design, configuration, or method of operation of the plant that could create the possibility of a new or different accident. Equipment will be operated in a manner for which it is currently designed. This license amendment request does not impact any plant systems that are accident initiators or adversely impact any accident mitigating systems. The Pressurizer Heaters are not themselves a credible initiator of any accident.

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

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

Response: No.

The proposed change would describe the specific common circuit breaker associated with the control power closing circuitry to the Switchgears 32A and 32B Supply Circuit Breakers in UFSAR 1.9.26 and 5.4.10 as contained in Attachment 2 [of the licensee's letter dated November 11, 2013] and that local manual operation outside of the Control Room would be necessary to reenergize Pressurizer Heaters during a loss of offsite power concurrent with the specific common circuit breaker being open. Plant Operators are trained and have procedural guidance including manual operator action to address Natural Circulation Cooldown with a Loss of Offsite Power.

This amendment does not change the manner in which safety limits or limiting safety settings are determined. Because the Pressurizer Heaters will continue to be monitored and controlled as per Technical Specification 3.4.3.1 and Technical Requirements Manual 3.4.3.1, this proposed change to the UFSAR will not present an adverse impact to plant operation or result in a significant reduction in a margin of safety.

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

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

Attorney for licensee: Joseph A. Aluise, Associate General Council—Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Douglas A. Broaddus.

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

Date of amendment request: December 9, 2013. A publicly-available version is in ADAMS under Accession No. ML13345A686.

Description of amendment request: Entergy Operations, Inc. (the licensee), has proposed to change the Waterford Steam Electric Station, Unit 3 Technical Specifications (TS). Specifically, the amendment would revise:

- TS 3.3.1, Reactor Protective Instrumentation;
- TS 3.1.3.4, Shutdown CEA [Control Element Assembly];
- TS 3.3.2, Engineered Safety Features Actuation System Instrumentation;
- TS 3.3.3.1, Radiation Monitoring Instrumentation;
- TS 3.3.3.6, Accident Monitoring Instrumentation;
- TS 3.3.3.11, Explosive Gas Monitoring Instrumentation;
- TS 4.8.2.1, D.C. [Direct Current] Sources;
- TS 6.1, Responsibility;
- TS 6.2.1, Offsite and Onsite Organizations;
- TS 6.2.2, Unit Staff; and
- TS 6.12, High Radiation Area.

These changes would improve clarity, correct administrative and typographical errors, or establish consistency with NUREG–1432, Standard Technical Specifications Combustion Engineering Plants, Revision 4.0 (NUREG–1432).

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

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

Response: No.

The proposed changes revise the Technical Specifications to improve clarity, correct administrative and typographical errors, and establish consistency with NUREG–1432. This includes two technical changes.

A provision to an existing surveillance test has been added that limits the total battery inter-cell resistance to maintain battery terminal voltage above the required operating voltage. A change to limit the total battery inter-cell resistance has no effect on the probability of an accident previously evaluated. The proposed change to limit the total battery inter-cell resistance does not involve a significant increase in the consequences of an accident previously evaluated. This is because the addition of this limit will ensure that the battery is demonstrated as capable to meet its safety function.

The other technical change extends the Completion Time from 1 hour to 4 hours for verifying that the departure from nucleate boiling ratio (DNBR) limit is met and disabling the Reactor Power Cutback when one or both CEACs [Control Element Assembly Calculators] are inoperable. A change to the Completion Time for Actions in response to inoperable equipment has no effect on the probability of an accident previously evaluated. The proposed change to the Completion Time for Actions in response to inoperable equipment does not involve a significant increase in the consequences of an accident previously evaluated. This is because the safety function of a CEAC is to identify and compensate for a misaligned CEA [control element assembly], and there is a low probability of occurrence during the four hour Completion Time that one or more misaligned CEAs could significantly adversely affect: Core power distribution, shutdown margin, ejected CEA worth, or initial reactivity insertion rate during a reactor trip.

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

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

Response: No.

The proposed changes revise the Technical Specifications to improve clarity, correct administrative and typographical errors, and establish consistency with NUREG–1432. This includes two technical changes.

A provision to an existing surveillance test has been added that limits the total battery inter-cell resistance to maintain battery terminal voltage above the required operating voltage. A change to limit the total battery inter-cell resistance does not create the possibility of a new or different kind of accident from any accident previously evaluated. This is because the addition of this limit will ensure that the battery is demonstrated as capable to meet its existing safety function and does not change the safety function in any manner.

The other technical change extends the Completion Time from 1 hour to 4 hours for verifying that the departure from nucleate boiling ratio (DNBR) limit is met and disabling the Reactor Power Cutback when one or both CEACs are inoperable. A change to the Completion Time for Actions in response to inoperable equipment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Consequently, the proposed changes do not create the possibility of a new or different kind of accident.

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

Response: No.

The proposed changes revise the Technical Specifications to improve clarity, correct administrative and typographical errors, and establish consistency with NUREG–1432. This includes two technical changes.

A provision to an existing surveillance test has been added that limits the total battery inter-cell resistance to maintain battery terminal voltage above the required operating voltage. A change to limit the total battery inter-cell resistance does not involve a significant reduction in a margin of safety. This is because the addition of this limit will ensure that the battery is demonstrated as having margin to meet its safety function.

The other technical change extends the Completion Time from 1 hour to 4 hours for verifying that the departure from nucleate boiling ratio (DNBR) limit is met and disabling the Reactor Power Cutback when one or both CEACs are inoperable. A change to the Completion Time for Actions in response to inoperable equipment does not affect protection criterion for plant equipment and does not reduce the margin of safety. This change provides Operators time to assess and perform the required activities in a controlled manner consistent with the risk associated with an inoperable CEAC function. Actions associated with this Condition involve disabling the Control Element Drive Mechanism Control System (CEDMCS), and signaling all OPERABLE CPC [core protection calculator] channels that both CEACs are failed. This applies a large penalty factor associated with two CEAC failures within CPC calculations. The penalty factor for two failed CEACs is sufficiently large that power must be maintained significantly <100% Reactor Thermal Power. The Completion Time of 4 hours is adequate to accomplish these actions while minimizing risks. Meeting the DNBR margin requirements ensures that power level and ASI [axial shape index] are within a conservative region of operation based on actual core conditions. In addition to the above actions, the Reactor Power Cutback System is disabled. This ensures that CEA position will not be affected by Reactor Power Cutback operation.

Consequently, there is no significant reduction in a margin of safety due to the proposed changes.

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

Attorney for licensee: Joseph A. Aluise, Associate General Council—Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, LA 70113.

NRC Branch Chief: Douglas A. Broaddus.

Exelon Generation Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, (TMI–1) Dauphin County, Pennsylvania

Date of amendment request: May 7, 2014. A publicly-available version is in ADAMS under Accession No. ML14127A424.

Description of amendment request: The amendment would change the TMI–1 technical specifications. Specifically, the proposed amendment would replace an existing Surveillance Requirement to operate ventilation systems with charcoal filters for a 10-hour period every 31 days with a requirement to operate the systems for greater than or equal to 15 continuous minutes every 31 days in accordance with Technical Specification Task Force (TSTF) Traveler TSTF–522, Revision 0, ''Revise Ventilation System Surveillance Requirements to Operate for 10 hours per Month'' (ADAMS Accession No. ML100890316).

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, along with NRC edits in square brackets:

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

Response: No.

The proposed change replaces an existing [Surveillance Requirement] SR to operate the Emergency Control Room Air Treatment System and the Fuel Handling Building [Engineered Safety Feature] ESF Air Treatment System for a 10-hour period at a frequency controlled in accordance with the [Surveillance Frequency Control Program] SFCP with a requirement to operate the systems for greater than or equal to 15 continuous minutes at a frequency controlled in accordance with the SFCP.

These systems are not accident initiators and therefore, these changes do not involve a significant increase in the probability of an accident. The proposed system and filter testing changes are consistent with current regulatory guidance for these systems and will continue to assure that these systems perform their design function, which may include mitigating accidents. Thus, the change does not involve a significant increase in the consequences of an accident.

Therefore, it is concluded that this 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 replaces an existing SR to operate the Emergency Control Room Air Treatment System and the Fuel Handling Building ESF Air Treatment System for a 10-hour period at a frequency controlled in accordance with the SFCP with a requirement to operate the systems for greater than or equal to 15 continuous minutes at a frequency controlled in accordance with the SFCP.

The change proposed for these ventilation systems does not change any system operations or maintenance activities. Testing requirements will be revised and will continue to demonstrate that the Limiting Conditions for Operation are met and the system components are capable of performing their intended safety functions. The change does not create new failure modes or mechanisms and no new accident precursors are generated.

Therefore, it is concluded that this 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 replaces an existing SR to operate the Emergency Control Room Air Treatment System and the Fuel Handling Building ESF Air Treatment System for a 10-hour period at a frequency controlled in accordance with the SFCP with a requirement to operate the systems for greater than or equal to 15 continuous minutes at a frequency controlled in accordance with the SFCP. The proposed change is consistent with regulatory guidance.

Therefore, it is concluded that this 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: J. Bradley Fewell, Esquire, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Acting Branch Chief: Robert G. Schaaf.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, (BVPS) Unit No. 1, Beaver County, Pennsylvania

Date of amendment request: July 30, 2013. A publicly-available version is in ADAMS under Accession No. ML13212A027.

Description of amendment request: The amendment would change the BVPS Facility Operating License. Specifically, the amendment requests authorization to implement 10 CFR 50.61a, ''Alternate fracture toughness requirements for protection against pressurized thermal shock events,'' in lieu of 10 CFR 50.61, ''Fracture toughness requirements for protection against pressurized thermal shock events.'' The 10 CFR 50.61 screening criteria define a limiting level of reactor pressure vessel embrittlement beyond which plant operation cannot continue without further evaluation. As described in NUREG–1806, ''Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61),'' the screening criteria in the PTS rule is overly conservative and the risk of through wall cracking due to a PTS event is much lower than previously estimated. A publicly-available version of NUREG–1806 is in ADAMS under Accession No. ML072830074.

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, with NRC edits in square brackets:

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

Response: No.

This amendment request would allow implementation of the alternate PTS [pressurized thermal shock] rule in lieu of 10 CFR 50.61 and would not involve a significant increase in the probability or consequences of an accident. Application of the alternate PTS rule in lieu of 10 CFR 50.61 would not result in physical alteration of a plant structure, system or component, or installation of new or different types of equipment. Further, application of the alternate PTS rule would not significantly affect the probability of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) or cause a change to any of the dose analyses associated with the UFSAR accidents because accident mitigation functions would remain unchanged. Use of the alternate PTS rule would change how fracture toughness of the reactor vessel is determined and does not affect reactor vessel neutron radiation fluence. As such, implementation of the alternate PTS rule in lieu of 10 CFR 50.61 would not increase the likelihood of a malfunction.

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 amendment request would allow implementation of the alternate PTS rule in lieu of 10 CFR 50.61. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. No physical plant alterations are made as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system.

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 amendment request would authorize implementation of the alternate PTS rule in lieu of 10 CFR 50.61. The alternate PTS rule would maintain the same functional requirements for the facility as 10 CFR 50.61. The alternate PTS rule establishes screening criteria that limit levels of embrittlement beyond which operation cannot continue without further plant-specific evaluation or modifications. Sufficient safety margins are maintained to ensure that any potential increases in core damage frequency and large early release frequency resulting from implementation of the alternate PTS rule are negligible. As such, there would be no significant reduction in the margin of safety as a result of use of the alternate PTS rule. The margin of safety associated with the acceptance criteria of accidents previously evaluated in the UFSAR is unchanged. The proposed change would have no affect on the availability, operability, or performance of the safety-related systems and components.

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

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

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

NRC Acting Branch Chief: Robert G. Schaaf.

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

Date of amendment request: April 16, 2014. A publicly-available version is in ADAMS under Accession No. ML14111A291.

Description of amendment request: The amendment would change BVPS–1 and BVPS–2 technical specifications (TSs). Specifically, the proposed license amendment would revise TS 5.5.12, ''Containment Leakage Rate Testing Program,'' Item a, by deleting reference to the BVPS–1 exemption letter dated December 5, 1984 (ADAMS Accession No. ML003766713), and requiring compliance with Nuclear Energy Institute (NEI) topical report NEI 94–01, Revision 3–A, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' (ADAMS Accession No. ML12221A202) instead of Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' (ADAMS Accession No. ML003740058) including listed exceptions. In summary, the amendment would allow extension of the Type A Reactor Containment Integrated Leak test, required by 10 CFR Part 50, Appendix J, interval to one test in 15 years and an extension of the Type C test interval to 75 months, based on acceptable performance history of the containment test as defined in NEI 94–01, Revision 3–A.

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, along with NRC edits in square brackets:

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

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94–01, Revision 3–A, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' for development of the Beaver Valley Power Station, Unit No. 1 (BVPS–1) and Unit No.2 (BVPS–2) performance-based containment testing program. NEI 94–01 allows, based on risk and performance, an extension of Type A and Type C containment leak test intervals. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses.

The findings of the Beaver Valley Power Station risk assessment confirm the general findings of previous studies that the risk impact with extending the containment leak rate is small. Per the guidance provided in Regulatory Guide 1.174, [An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (ADAMS Accession No. ML100910006)] [* * *] an extension of the leak test interval in accordance with NEI 94–01 [Revision 3–A] results in an estimated change within the very small change region.

Since the change is implementing a performance-based containment testing program, the proposed amendment does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The requirement for leakage rate acceptance will not be changed by this amendment. Therefore, the containment will continue to perform its design function as a barrier to fission product releases.

Therefore, the proposed amendment does not significantly increase 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 to implement a performance-based containment testing program, associated with integrated leakage rate test frequency, does not change the design or operation of structures, systems, or components of the plant. In addition, the proposed changes would not impact any other plant system or component.

The proposed changes would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses. There are no accident initiators created or affected by these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. [* * *]

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

Response: No.

The proposed change to implement a performance-based containment testing program, associated with integrated leakage rate test frequency, does not affect plant operations, design functions, or any analysis that verifies the capability of a structure, system, or component of the plant to perform a design function. In addition, this change does not affect safety limits, limiting safety system setpoints, or limiting conditions for operation.

The specific requirements and conditions of the Technical Specification Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by Technical Specifications is maintained. This ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by implementation of a performance-based containment testing program.

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

NRC Acting Branch Chief: Robert G. Schaaf.

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

Date of amendment request: June 2, 2014. A publicly-available version is in ADAMS under Accession No. ML14157A006.

Description of amendment request: The proposed amendment would revise the Cooper Nuclear Station Technical Specifications (TS) to update Figure 4.1–1, ''Site and Exclusion Area Boundaries and Low Population Zone,'' to reflect the current site layout.

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 updates a figure with the current site layout. An administrative change such as this is not an initiator of any accident previously evaluated. As a result, the probability of an accident previously evaluated is not affected. The consequences of an accident with the incorporation of this administrative change are not different than the consequences of the same accident without this change. As a result, the consequences of an accident previously evaluated are not affected by this change.

Based on the above, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not modify the plant design, nor does the proposed change alter the operation of the plant or equipment involved in either routine plant operation or in the mitigation of design basis accidents. The proposed change is administrative only.

Based on the above, it is concluded that 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 amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change consists of an administrative change to update a figure of the site layout. The change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside of the design basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: November 14, 2013. A publicly-available version is in the Agencywide Documents Access and Management System under Accession No. ML13322A446.

Description of amendment request: NSPM proposes to revise the MNGP technical specification (TS) 5.5.11, ''Primary Containment Leakage Rate Testing Program,'' airlock testing conditions. Specifically, NSPM proposes to remove the reduced pressure testing option for drywell airlock door leakage testing in accordance with the requirements of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR 50), Appendix J, Option B, since this capability is not required and does not reflect the current testing practice at MNGP.

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 provided 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 the TS allowance to test the leakage rate of the drywell personnel airlock doors at a reduced pressure. However, overall airlock leakage rate testing will continue to be performed in accordance with Option B of 10 CFR 50, Appendix J. Removal of this capability does not affect, nor is it a precursor for, an accident or transient analyzed in the MNGP Updated Safety Analysis Report. The proposed change does not change the total allowable primary containment leakage rate, nor does it involve a change to the physical design and operation of the plant.

Therefore, operation of the facility in accordance with 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 the TS allowance to test the leakage rate of the drywell personnel airlock doors at a reduced pressure. However, overall airlock leakage rate testing will continue to be performed in accordance with Option B to 10 CFR 50, Appendix J. The change being proposed will not change the physical plant or modes of operation defined in the facility license. The proposed change does not increase the total allowable primary containment leakage rate. The change does not involve the addition or modification of equipment, nor does it alter the design or operation of plant systems.

Therefore, operation of the facility in accordance with 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 the TS allowance to test the leakage rate of the drywell personnel airlock doors at a reduced pressure. However, overall airlock leakage rate testing will continue to be performed in accordance with Option B to 10 CFR 50, Appendix J. The proposed change does not affect plant safety analyses or change the physical design or operation of the plant. The proposed change does not increase the total allowable primary containment leakage rate.

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

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

Attorney for licensee: Peter M. Glass, Assistant General Counsel, Xcel Energy Services, Inc., 414 Nicollet Mall, Minneapolis, MN 55401.

NRC Branch Chief: David L. Pelton.

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 amendment request: June 9, 2014. A publicly-available version is in ADAMS under Accession No. ML14160A593.

Description of amendment request: The proposed amendments would revise the Prairie Island Nuclear Generating Plant, Units 1 and 2, Surveillance Requirements 3.8.1.2, 3.8.1.6, and 3.8.1.9 associated with steady state voltage and frequency limits in Technical Specification3.8.1, ''AC [Alternating Current] Sources—Operating.''

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

This license amendment request proposes to revise specific emergency diesel generator steady states voltage and frequency limits in the Technical Specification Surveillance Requirements which are more restrictive than the current limits.

The emergency diesel generators and the equipment on the safeguards buses supplied by the emergency diesel generators are not accident initiators, and therefore the proposed voltage and frequency limits changes do not involve an increase in the probability of an accident.

The proposed emergency diesel generator surveillance test voltage and frequency limits assure the emergency diesel generators are capable of providing electrical power at voltages and frequencies that are adequate to operate the required equipment on the safeguards buses and thus maintain the current licensing basis for accident mitigation. Thus the proposed voltage and frequency limit changes do not involve a significant increase in the consequences of an accident.

Therefore, the proposed Technical Specification 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

This license amendment request proposes to revise specific emergency diesel generator steady state voltage and frequency limits in the Technical Specification Surveillance Requirements which are more restrictive than the current limits.

The proposed Technical Specification changes which revise the emergency diesel generator voltage and frequency limits do not change any system operations or maintenance activities. The changes do not involve physical alteration of the plant; that is, no new or different type of equipment will be installed. The changes do not alter assumptions made in the safety analyses but ensure that the diesel generators are capable of operating equipment as assumed in the accident analyses. These changes do not create new failure modes or mechanisms which are not identifiable during testing and no new accident precursors are generated.

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

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

Response: No

This license amendment request proposes to revise specific emergency diesel generator steady state voltage and frequency limits in the Technical Specification Surveillance Requirements which are more restrictive than the current limits.

Since this license amendment proposes Technical Specification changes which further restrict the acceptable voltage and frequency limits, both upper and lower, margins of safety are increased, and no margin of safety is reduced as part of this change.

Therefore, the proposed Technical Specification 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 requests involve no significant hazards consideration.

Attorney for licensee: Peter M. Glass, Assistant General Counsel, Xcel Energy Services, Inc., 414 Nicollet Mall, Minneapolis, MN 55401

NRC Branch Chief: David L. Pelton.

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

Date of amendment request: March 19, 2014. A publicly-available version is in ADAMS under Accession No. ML14079A599.

Description of amendment request: The requested amendment reclassifies portions of the five Tier 2* Human Factors (HF) Verification & Validation (V&V) planning documents listed in the Updated Final Safety Analysis Report (UFSAR) Table 1.6–1 and Chapter 18, Subsection 18.11.2. These five documents outline the overall plan for the HF V&V, including the Human Factors Engineering (HFE) design verification, task support verification, integrated system validation, discrepancy resolution process, and verification at plant startup. The licensee stated that the requested amendment identifies the portions of the five HF V&V planning documents that would more appropriately be classified as Tier 2, due to those portions having no impact on safety, and proposes the necessary departures to reclassify this information. This differentiation between Tier 2 and Tier 2* information in the HF V&V planning documents will allow for revisions of these documents using the Tier 2 change process provided in 10 CFR Part 52 Appendix D, Section VIII.B.5. Because this proposed change requires a departure from Tier 2* 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 2* in accordance with 10 CFR Part 52 Appendix D Section VIII B.6.c.(15).

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 changes reclassify portions of the five Tier 2* Human Factors (HF) Verification & Validation (V&V) planning documents listed in the Updated Final Safety Analysis Report (UFSAR). These changes do not modify the design, construction, or operation of any plant structures, systems, or components (SSC), nor do they change any procedures or method of control for any SSCs. Because the proposed changes do not change the design, construction, or operation of any SSCs, they do not adversely affect any design function as described in the UFSAR. Therefore, the proposed amendment does not affect the probability of an accident previously evaluated. Similarly, because the proposed changes do not alter the design or operation of the nuclear plant or any plant SSCs, the proposed changes do not represent a change to the radiological effects of an accident, and therefore, they do not involve an increase in the consequences of an accident previously evaluated.

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 not a modification, addition to, or removal of any plant SSCs. Furthermore, the proposed changes are not a change to procedures or method of control of the nuclear plant or any plant SSCs. The only impact of this activity is the reclassification of portions of the five HF V&V planning documents as Tier 2 information. Because the proposed amendment does not change the design, construction, or operation of the nuclear plant or any plant operations, it does not affect the possibility of an 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 a margin of safety?

Response: No

The proposed changes reclassify portions of the five Tier 2* HF V&V planning documents listed in the UFSAR from Tier 2* to Tier 2. The proposed amendment only affects the classification of planning documents and does not change the design, construction, or operation of the nuclear plant or any plant operations; therefore, the changes do not affect any margin of safety.

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

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

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

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: July 3, 2014. A publicly available version is available in the Agencywide Documents Access and Management System under Accession No. ML14187A533.

Description of amendment request: The purpose of the proposed license amendment request is to address proposed changes related to the design details of the containment internal structural wall modules (CA01, CA02, and CA05). The proposed changes to Tier 2 information in the Updated Final Safety Analysis Report (UFSAR), and the involved plant-specific Tier 1 and corresponding combined license Appendix C information would allow the use of thicker than normal faceplates to accommodate local demand or connection loads in certain areas without the use of overlay plates or additional backup structures. Additional proposed changes to Tier 2 information and involved Tier 2* information would allow:

(1) A means of connecting the structural wall modules to the base concrete via use of structural shapes, reinforcement bars, and shear studs extending horizontally from the structural module faceplates and embedded during concrete placement as an alternative to the use of embedment plates and vertically oriented reinforcement bars,

(2) A variance in structural module wall thicknesses from the thicknesses identified in UFSAR Figure 3.8.3–8, ''Structural Modules—Typical Design Details,'' for some walls that separate equipment spaces from personnel access areas, and

(3) The use of steel plates, structural shapes, reinforcement bars, or tie bars between the module faceplates, as needed to support localized loads and ensure compliance with applicable codes.

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

Response: No

The design function of the internal containment structures is to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in those structures. These structures are structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29.

The changes to the design details for the structural modules 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, nor do they change the seismic Category I classification.

Evaluations have been performed which determined that the proposed changes do not have a significant impact on the calculated loads for the affected structural modules, or critical locations, and no significant impact on the global seismic model. The changes to the design details for the structural modules 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 does the change described create any new accident precursors.

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

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

Response: No.

The proposed changes are to revise design details for the internal containment structural modules. The changes do not change the design requirements of the nuclear island structures, nor do they change the seismic Category I classification. The changes to the design details for the internal containment structural modules do not change the design function, support, design, or operation of mechanical and fluid systems. The changes to the design details for the internal containment structural modules do 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 requested 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 requested amendment proposes changes to the structural details associated with the in-containment structural modules. The purpose of these changes is to ensure that the requirements contained in the applicable construction codes are met. As discussed in UFSAR, Section 3.8.3.5, ''Design Procedures and Acceptance Criteria,'' the in-containment structural modules are designed in accordance with ACI 349 and AISC N690. Thus, the identification of additional structural module connection details, the increase in structural module faceplate and wall thicknesses, and the addition of additional reinforcement in specific areas are proposed to ensure that the codes of record, and the associated margins contained therein, continue to be met as specified in the design basis. Structural and seismic analysis of the modified sections in accordance with the methodologies identified in the UFSAR has confirmed that the applicable requirements of ACI 349 and AISC N690 continue to be met for affected in-containment structural modules.

As a result, the proposed changes do not adversely affect any safety-related equipment or other design functions, design code compliance, design analysis, safety analysis input or result, or design/safety margin. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes. Therefore, the requested 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: M. Stanford Blanton, Blach & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203–2015.

NRC Branch Chief: Lawrence Burkhart.

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

Date of amendment request: January 6, 2014, as supplemented by letter dated June 9, 2014. Publicly-available versions are in ADAMS under Accession Nos. ML14035A075 and ML14184B363.

Description of amendment request: The proposed license amendment would revise Technical Specification (TS) 3.3.1, ''Reactor Trip System Instrumentation,'' with respect to the required actions and allowed outage times for inoperable reactor trip breakers. The proposed changes would revise the required actions to enhance plant reliability by reducing exposure to unnecessary shutdowns and increase operational flexibility by allowing more time to make required repairs for inoperable reactor trip breakers consistent with allowed outage times for associated logic trains. No modifications to setpoint actuations, trip setpoint, surveillance requirements or channel response that would affect the safety analyses are associated with the proposed changes.

The proposed changes are consistent with requirements generically approved as part of NUREG–1431, Standard Technical Specifications, Westinghouse Plants, Revision 4 (TS 3.3.1, ''Reactor Trip System Instrumentation''). Justification for the proposed changes is based on Westinghouse Electric Company LLC's topical report WCAP–15376–P–A, Revision 1, ''Risk-Informed Assessment of the RTS [Reactor Trip System] and ESFAS [Engineered Safety Feature Actuation System] Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,'' March 2003 (not publicly available; proprietary).

This application was originally noticed in the **Federal Register** on April 8, 2014 (79 FR 19400), as a license amendment request containing sensitive unclassified non-safeguards information (SUNSI). However, by letter dated June 9, 2014, STP Nuclear Operating Company removed all proprietary markings from Attachment A of Enclosure 1, ''Topical Report Applicability Determination, ST–WN–NOC–13–46,'' originally included in the letter dated January 6, 2014. Therefore, the application is being renoticed in the **Federal Register** to remove the SUNSI designation.

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 overall reactor trip breaker performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The reactor trip breakers will continue to function in a manner consistent with the plant design basis.

The proposed changes do not introduce any new accident initiators, and therefore do not increase the probability of any accident previously evaluated. There will be no degradation in the performance of or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the Updated Final Safety Analysis Report.

The determination that the results of the proposed changes are acceptable was established in the NRC Safety Evaluation (issued by letter dated December 20, 2002) prepared for WCAP–15376–P–A, ''Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times'' [ADAMS Accession No. ML023540534]. Implementation of the proposed changes will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the respective NRC Safety Evaluation conditions.

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

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

Response: No.

The proposed changes do not result in a change in the manner in which the Reactor Trip Breakers provide plant protection. The proposed changes do not change the response of the plant to any accidents. No design changes are associated with 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. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

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

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

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria as stated in the Updated Final Safety Analysis Report are not impacted by these changes. Redundant Reactor Trip Breaker features and diverse trip features for each Reactor Trip Breaker are maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses are unaffected by the proposed change. The proposed changes will not result in plant operation in a configuration outside the design basis. The proposed changes should enhance plant reliability by reducing exposure to unnecessary shutdowns and increase operational flexibility by allowing more time to make required repairs for inoperable reactor trip breakers. The calculated impact on risk is insignificant and meets the acceptance criteria contained in NRC Regulatory Guides 1.174 [''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' Revision 2 (ADAMS Accession No. ML100910006)] and 1.177 [''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' Revision 1 (ADAMS Accession No. ML100910008)].

Therefore, the proposed changes do 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: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue NW., Washington, DC 20004.

NRC Branch Chief: Michael T. Markley.

III. 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 can be accessed as described in the ''Obtaining Information and Submitting Comments'' section of this document.

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

Date of amendment request: May 3, 2013, as supplemented by letters dated July 2 and October 2, 2013, and January 15 and May 28, 2014.

Brief description of amendment: The amendment revised the Technical Specifications.

Date of issuance: July 10, 2014.

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

Amendment No.: 260. A publicly-available version is in ADAMS under Accession No. ML14178A599; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF–49: Amendment revised the Renewed Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register:*** August 20, 2013 (78 FR 51225). The supplemental letters dated July 2 and October 2, 2013, and January 15 and May 28, 2014, 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 July 10, 2014.

No significant hazards consideration comments received: No.

Duke Energy Florida, Inc., et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit 3, Citrus County, Florida

Date of amendment request: April 25, 2013, as supplemented by letters dated September 4, 2013, and February 26, 2014.

Brief description of amendment: The amendment revised and removed certain requirements from the Section 5.0, ''Administrative Controls,'' portions of the Technical Specifications (TSs) that are no longer applicable to the facility in its permanently shutdown and defueled condition.

Date of issuance: July 11, 2014.

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

Amendment No.: 244. A publicly-available version is in ADAMS under Accession No. ML14097A145; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

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

*Date of initial notice in **Federal Register:*** July 23, 2013 (78 FR 44174). The supplemental letter dated September 4, 2013, expanded the scope of the application as originally noticed; therefore, the staff re-noticed the application and included a revised proposed no significant hazards consideration determination on November 12, 2013 (78 FR 67406). The supplemental letter dated February 26, 2014, provided additional information that clarified the supplement dated September 4, 2013, did not expand the scope of the application as noticed on November 12, 2013, and did not change the NRC staff's proposed no significant hazards consideration determination as published in the **Federal Register** on November 12, 2013.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 2014.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: January 28, 2013, as supplemented by letters dated August 21, 2013, and April 22, 2014.

Brief description of amendment(s): Nuclear Safety Advisory Letter 11–5 identified Westinghouse methodology errors in the long-term mass and energy releases during a large break loss-of-coolant accident. These impacted the containment integrity analysis for Indian Point Unit No. 3 and required revisions to the limiting initial operating conditions (i.e., containment temperature, containment pressure, and refueling water storage tank temperature) and required revisions to Technical Specifications (TSs) 3.5.4, ''Refueling Water Storage Tank (RWST),'' and 3.6.4, ''Containment Pressure.'' In addition, revisions were made to TS 3.6.3, ''Containment Isolation Valves,'' to delete a redundant surveillance requirement and TS 5.5.15, ''Containment Leakage Rate Testing

Program,'' to reflect a slightly higher calculated containment peak pressure.

Date of issuance: July 17, 2014.

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

Amendment No.: 253. A publicly-available version is in ADAMS under Accession No. ML14169A583; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment(s).

Facility Operating License No. DPR–64: The amendment revised the Facility Operating License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 2, 2013 (78 FR 19750). The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC 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 July 17, 2014.

No significant hazards consideration comments received: No.

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

Date of amendment request: January 28, 2013, as supplemented by letters dated August 21, 2013, and April 22, 2014.

Brief description of amendment(s): The amendment authorizes revisions to the Indian Point Unit No. 2 Updated Final Safety Analysis Report (UFSAR) to credit four rather than three containment fan cooler units in the containment integrity analysis. A re-analysis of the large break loss-of-coolant accident was performed to correct methodology errors in the long-term mass and energy releases for the containment integrity analysis and crediting four containment fan cooler units for the limiting single failure is necessary to maintain the peak containment pressure within the current analysis of record.

Date of issuance: July 16, 2014.

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

Amendment No.: 276. A publicly-available version is in ADAMS under Accession No. ML14126A809; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

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

Date of initial notice in ***Federal Register:*** April 2, 2013 (78 FR 19749). The supplement letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register.**

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 16, 2014.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant (JAFNPP), Oswego County, New York

Date of application for amendment: May 7, 2013, as supplemented by letter dated January 17, 2014.

Brief description of amendment: The amendment revised License Condition 2.T of the JAFNPP Renewed Facility Operating License to be consistent with the license condition contained in NUREG–1905, ''Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant,'' dated April 2008, and to clarify that the programs and activities described in the Updated Final Safety Analysis Report Supplement and identified in Appendix A of NUREG–1905 are to be completed no later than the start of the period of extended operation (PEO). The change removes any potential inference that any of the activities are being implemented after the PEO begins.

Date of issuance: July 16, 2014.

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

Amendment No.: 306. A publicly-available version is in ADAMS under Accession No. ML14086A152, documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR–59: The amendment revised the License.

Date of initial notice in ***Federal Register:*** April 15, 2014 (79 FR 21297).

The January 17, 2014, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 16, 2014.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 23, 2012, as supplemented by letter dated May 1, 2013. Publicly-available versions are in ADAMS under Accession Nos. ML12206A057 and ML13122A046, respectively.

Description of amendment: The amendments delete the limiting condition for operation Note associated with technical specifications (TS) Section 3.5.3, ''ECCS—Shutdown.''

Date of issuance: July 21, 2014.

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

Amendment Nos.: 176/182. A publicly-available version is in ADAMS under Accession No. ML13311B481; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

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

Date of initial notice in ***Federal Register:*** (77 FR 67682), dated November 13, 2012.

The supplement letter dated May 1, 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 as published in the **Federal Register.**

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 2014.

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

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

Date of application for amendment: December 21, 2012.

Brief description of amendment:

The proposed amendment would revise Technical Specification (TS) 3.3.6, ''Containment Ventilation Isolation Instrumentation.'' Specifically, this amendment request proposes to revise Footnote (b) of TS Table 3.3.6–1, ''Containment Ventilation Isolation Instrumentation,'' which specifies the ''Containment Radiation—High'' trip setpoint for two containment area radiation monitors (i.e., 1(2)RE–AR011 and 1(2)RE–AR012). The proposed changes would revise the ''Containment Radiation—High'' trip setpoint from the current, overly conservative value (i.e., a submersion dose rate of less than or equal to 10 milliroentgen per hour (mR/hr) in the containment building), to less than or equal to 2 times the containment building background radiation reading at rated thermal power, which is consistent with NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.'' Upon reaching the ''Containment Radiation—High'' setpoint, these area radiation monitors provide an isolation signal to the containment normal purge, minipurge, and post-loss of coolant accident systems' containment isolation valves.

Date of issuance: July 21, 2014.

Effective date: As of the date of issuance and shall be implemented within 165 days.

Amendment Nos.: 178/178; 184/184. (ADAMS Accession No. ML14106A169; documents related to these amendments are in the Safety Evaluation referenced in this notice).

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

Date of initial notice in ***Federal Register:*** (78 FR 22568), dated April 16, 2013.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 2014.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

Date of amendment request: September 3, 2013, (ADAMS Accession No. ML13246A321).

Brief description of amendments:

The amendments modify technical specifications (TSs) requirements to operate ventilation systems with charcoal filters for 10 hours, at a frequency specified in the Surveillance Frequency Control Program, in accordance with Technical Specification Task Force (TSTF)–522, Revision 0, ''Revise Ventilation System Surveillance Requirements to Operate for 10 hours per Month.'' A notice of the availability of TSTF–522 and a model safety evaluation was published in the **Federal Register** on September 20, 2012 (77 FR 58421).

Date of issuance: July 21, 2014.

Effective date: As of the date of issuance and shall be implemented within 105 days.

Amendment Nos.: 177/177; 183/183; 201; 241/234; 208/195; 252/247. A publicly-available version is in ADAMS under Accession No. ML14085A532; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. NPF–72, NPF–77, NPF–37, NPF–66, NPF–62, DPR–19, DPR–25, NPF–11, NPF–18, DPR–29, and DPR–30: The amendments revised the TSs and Licenses.

Date of initial notice in ***Federal Register:*** December 24, 2013 (78 FR 77732).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 2014.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 28th day of July 2014.

For the Nuclear Regulatory Commission.

A. Louise Lund,

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

[FR Doc. 2014–18395 Filed 8–4–14; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2014–0168]

Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information

AGENCY: Nuclear Regulatory Commission.

ACTION: License amendment request; opportunity to comment, request a hearing, and petition for leave to intervene; order.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) received and is considering approval of seven amendment requests. The amendment requests are for James A. Fitzpatrick Nuclear Power Plant; Pilgrim Nuclear Power Station; Calvert Cliffs Nuclear Power Plant; LaSalle County Station, Units 1 and 2 (two requests); Nine Mile Point Nuclear Station, Unit 2; Prairie Island Nuclear Power Plant, Units 1 and 2. For each amendment request, the NRC proposes to determine that they involve no significant hazards consideration. In addition, each amendment request contains sensitive unclassified non-safeguards information (SUNSI).

DATES: Comments must be filed by September 4, 2014. A request for a hearing must be filed by October 6, 2014. Any potential party as defined in § 2.4 of Title 10 of the *Code of Federal Regulations* (10 CFR), who believes access to SUNSI is necessary to respond to this notice must request document access by August 15, 2014.

ADDRESSES: You may submit comments 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–2014–0168. Address questions about NRC dockets to Carol Gallagher; telephone: 301–287–3422; email: *Carol.Gallagher@nrc.gov.* For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.
- *Mail comments to:* Cindy Bladey, Office of Administration, Mail Stop: 3WFN–06–A44M, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.