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[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2013-0045]

Biweekly Notice

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

Background

Pursuant to Section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license 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 February 7, 2013, to February 20, 2013. The last biweekly notice was published on February 19, 2013 (78 FR 11688).

ADDRESSES: You may access information and comment submissions related to this document, which the NRC possesses and is publicly available, by searching on http://www.regulations.gov under Docket ID **<NRC-20YY-XXXX>**. You may submit comments by the following methods:

- **Federal rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID <**NRC-20YY-XXXX**>. Address questions about NRC dockets to Carol Gallagher; telephone: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.
- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.
- **Fax comments to:** RADB at 301-492-3446.

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

SUPPLEMENTARY INFORMATION:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID <**NRC-20YY-XXXX**> when contacting the NRC about the availability of information regarding this document. You may access information related to this document, which the NRC possesses and is publicly available, by the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID <**NRC-20YY-XXXX**>.
- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly available documents online in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public

Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. Documents may be viewed in ADAMS by performing a search on the document date and docket number.

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

B. Submitting Comments

Please include Docket ID **<NRC-20YY-XXXX>** 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. The NRC posts all comment submissions at http://www.regulations.gov as well as entering the comment submissions into ADAMS, and the NRC does not 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 in their comment submissions that they do not want to be publicly disclosed. Your request should state that the NRC will not edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

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

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

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

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

Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

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

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

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

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, then any hearing held would take place before the issuance of any amendment.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

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

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

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

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

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at 1-866 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at http://ehd1.nrc.gov/ehd/, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. 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 following three factors in 10 CFR 2.309(c)(1): (i) the information upon which the filing is based was not previously available; (ii) the information upon which the filing is based is materially different from information previously available; and (iii) the filing has been submitted in a timely fashion based on the availability of the subsequent information.

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

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendment request: December 12, 2012.

Description of amendment request: The amendments would change the Technical Specifications (TSs) by replacing the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on DOSE EQUIVALENT XE-133 and would take into account only the noble gas activity in the primary coolant. The changes are consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-490, Revision 0, "Deletion of E-Bar Definition and Revision to RCS [Reactor Coolant System] Specific Activity Technical Specifications," with deviations.

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. The license concluded that the no significant hazards consideration determination published in the *Federal Register* on March 19, 2007 (72 FR 12838), is applicable, and is presented below:

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

   Response: Reactor coolant specific activity is not an initiator for any accident previously evaluated. The Completion Time when primary coolant gross activity is not within limit is not an initiator for any accident previously evaluated. The current variable limit on primary coolant iodine concentration is not an initiator to any accident previously evaluated. As a result, the proposed change does not significantly increase the probability of an accident. The proposed change will limit primary coolant noble gases to concentrations consistent with the accident analyses. The proposed change to the Completion Time has no impact on the consequences of any design basis accident since the consequences of an accident during the extended Completion Time are the same as the consequences of an accident during the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

   Response: The proposed change in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter. The change does not create the potential for a new or different kind of accident from any previously calculated.

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

   Response: The proposed change revises the limits on noble gase [sic] radioactivity in the primary coolant. The proposed change is consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses.

   Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

Attorney for licensee: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072-2034.

NRC Branch Chief: Michael T. Markley.

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendment request: December 26, 2012.

Description of amendment request: The amendments would adopt Technical Specifications Task Force (TSTF) Traveler TSTF-500, Revision 2, “DC Electrical Rewrite - Update to TSTF-360,” with one variation. The amendments would revise the TS requirements related to direct current (DC) electrical systems in TS Limiting Condition for Operation (LCO) 3.8.4, “DC Sources - Operating,” LCO 3.8.5, “DC Sources - Shutdown,” and LCO 3.8.6, “Battery Parameters.” In addition, new TS 5.5.19, “Battery Monitoring and Maintenance Program,” is being proposed for Section 5.5, “Administrative Controls - Programs and Manuals.”

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 any accident previously evaluated?

   Response: No.

   The proposed changes restructure the Technical Specifications (TS) for the direct current (DC) electrical power system and are consistent with TSTF-500, Revision 2. The proposed changes modify TS Actions relating to battery and battery charger inoperability. The DC electrical power system, including associated battery chargers, is not an initiator of any accident sequence analyzed in the Updated Final Safety Analysis Report (UFSAR). Rather, the DC electrical power system supports equipment used to mitigate accidents. The proposed changes to restructure TS and change surveillances for batteries and chargers to incorporate the updates included in TSTF-500, Revision 2, will maintain the same level of equipment performance required for mitigating accidents assumed in the UFSAR. Operation in accordance with the proposed TS would ensure that the DC electrical power system is capable of performing its specified safety function as described in the UFSAR. Therefore, the mitigating functions supported by the DC electrical power system will continue to provide the protection assumed by the analysis. The relocation of preventive maintenance surveillances, and certain operating limits and actions, to a licensee-controlled *Battery Monitoring and Maintenance Program* will not challenge the ability of the DC electrical power system to perform its design function. Appropriate monitoring and maintenance that are consistent with industry standards will continue to be performed. In addition, the DC electrical power system is within the scope of 10 CFR 50.65, *Requirements for monitoring the effectiveness of maintenance at* *nuclear power plants*, which will ensure the control of maintenance activities associated with the DC electrical power system.

The integrity of fission product barriers, plant configuration, and operating procedures as described in the UFSAR will not be affected by the proposed changes. Therefore, the consequences of previously analyzed accidents will not increase by implementing these changes. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes involve restructuring the TS for the DC electrical power system. The DC electrical power system, including associated battery chargers, is not an initiator to any accident sequence analyzed in the UFSAR. Rather, the DC electrical power system supports equipment used to mitigate accidents. The proposed changes to restructure the TS and change surveillances for batteries and chargers to incorporate the updates included in TSTF-500, Revision 2, will maintain the same level of equipment performance required for mitigating accidents assumed in the UFSAR. Administrative and mechanical controls are in place to ensure the design and operation of the DC systems continues to meet the plant design basis described in the UFSAR. Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The equipment margins will be maintained in accordance with the plant-specific design bases as a result of the proposed changes. The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC capacity to support operation of mitigation equipment is ensured. The changes associated with the new *Battery Maintenance and Monitoring Program* will ensure that the station batteries are maintained in a highly reliable manner. The equipment fed by the DC electrical sources will continue to provide adequate power to safety-related loads in accordance with analysis assumptions.

> TS changes made in accordance with TSTF-500, Revision 2, maintain the same level of equipment performance stated in the UFSAR and the current TSs. Therefore, the proposed changes do not involve a significant reduction of safety.

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

Attorney for licensee: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072-2034.

NRC Branch Chief: Michael T. Markley.

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

Date of amendments request: October 2, 2012, as supplemented by letter dated November 26, 2012.

Description of amendments request: The amendments would revise Technical Specification (TS) 3.8.3 “Diesel Fuel Oil” by relocating the current stored diesel fuel oil numerical volume requirements from the TS to the TS Bases and TS 3.8.1 “AC Sources-Operating” by relocating the specific numerical value for the day tank fuel oil volume from the TS to the TS Bases. The changes would be consistent with Nuclear Regulatory Commission (NRC)-approved Industry Technical Specification Task Force Standard Technical Specification Change Traveler, TSTF-501-A, Revision 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. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

   No.

   The proposed change relocates the volume of diesel fuel oil required to support 7-day operation of an onsite diesel generator, and the volume equivalent to a 6-day supply, to licensee control. The specific volume of fuel oil equivalent to a 7- and 6-day supply is calculated using the limiting energy content of the fuel, the required diesel generator output and the corresponding fuel oil consumption rate. Because the requirement to maintain a 7-day supply of diesel fuel oil is not changed and is consistent with the assumptions in the accident analysis, and the actions taken with the volume of fuel oil is less than a 6-day supply have not changed, neither the probability nor the consequences of any accident previously evaluated will be affected.

   The proposed change also relocates the volume of diesel fuel oil required to support one hour of diesel generator operation at full load in the day tank. The specific volume and time is not changed and is consistent with the existing plant design basis to support a diesel generator under accident load conditions.

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

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

   No.

   The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis but ensures that the diesel generator operates as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

   The proposed change also relocates the volume of diesel fuel oil required to support one hour of diesel generator operation at full load in the day tank. The change does not alter assumptions made in the safety analysis but ensures that the diesel generator operates as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No.

The proposed change relocates the volume of diesel fuel oil required to support 7-day operation of an onsite diesel generator, and the volume equivalent to a 6-day supply, and one hour day tank supply to licensee control. As the basis for the existing limits on diesel fuel oil are not changed, no change is made to the accident analysis assumptions and no margin of safety is reduced as part of this change.

The proposed change also relocates the volume of diesel fuel oil required to support one hour of diesel generator operation at full load in the day tank. As the basis for the existing limits on diesel fuel oil are not changed, no change is made to the accident analysis assumptions and no margin of safety is reduced as part of this change.

Therefore, the proposed amendment would 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: Steven L. Miller, General Counsel, Constellation Energy Nuclear Group, LLC, 100 Constellation Way, Suite 200c, Baltimore, MD 21202.

NRC Branch Chief: George Wilson.

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

Date of amendments request: October 16, 2012.

Description of amendments request: The amendments would revise Surveillance Requirements (SRs) 3.8.1.8, 3.8.1.11, and 3.8.2.1 and add SR 3.8.1.17 of Technical Specification (TS) 3.8.1 "AC 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 amendment request proposes to add or modify certain [TS SRs] for the diesel generators. This proposed amendment will provide additional assurance that the AC Sources relied upon to ensure the availability of necessary power to the Engineered Safety Features systems are capable of performing their specified safety function if needed. The diesel generators and their associated emergency loads are accident mitigating features, not accident initiators. This proposed amendment does not change the design function of the diesel generators or any of their required loads, and does not change the way the systems and plant are operated or maintained. This proposed amendment does not impact any plant systems that are accident initiators and does not adversely impact any accident mitigating systems.

   The proposed amendment does not affect the operability requirements for the diesel generators, as verification of such operability will continue to be performed as required. Continued verification of operability supports the capability of the diesel generators to perform their required design functions of providing emergency power to the Engineered Safety Features systems, consistent with the plant safety analyses as described in the Updated Final Safety Analysis Report (UFSAR).

   Adding or modifying [TS SRs] for the diesel generators will not significantly increase the probability of an accident previously evaluated because the diesel generators and their emergency loads are accident mitigation features, not accident initiators. Adding or modifying [TS SRs] for the diesel generators will not change any of the dose analyses associated with the UFSAR Chapter 14 accidents because accident mitigation functions and requirements remain unchanged.

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.

   This amendment request proposes to add or modify certain [TSs SRs] for the diesel generators. This proposed amendment does not change the design function of the diesel generators or any required loads, and does not change the way the systems and plant are operated or maintained. This proposed amendment does not impact any plant systems that are accident initiators and does not adversely impact any accident mitigating systems. Performance of these surveillances tests will provide additional assurance that the AC Sources relied upon to ensure the availability of necessary power to the Engineered Safety Features systems are capable of performing their specified safety function if needed.

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

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

   Response: No.

   This amendment request proposes to add or modify certain [TS SRs] for the diesel generators. This proposed amendment will provide additional assurance that the AC Sources relied upon to ensure the availability of necessary power to the Engineered Safety Features systems are capable of performing their specified safety function if needed. Margin of safety is related to the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. This proposed amendment does not involve or affect fuel cladding, the reactor coolant system, or the primary containment. Performance of these surveillances tests will provide continued assurance that the AC Sources relied upon to ensure the availability of necessary power to the Engineered Safety Features systems are capable of performing their specified safety function if needed.

   Therefore, the proposed amendment 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: Steven L. Miller, General Counsel, Constellation Energy Nuclear Group, LLC, 100 Constellation Way, Suite 200c, Baltimore, MD 21202

NRC Branch Chief: George Wilson.

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

Date of amendment request: December 21, 2012.

Description of amendment request: The proposed amendment would revise the Fermi 2 operating license to change its name on the license to "DTE Electric Company." This name change is purely administrative in nature. Detroit Edison is a wholly owned subsidiary of DTE Energy Company, and this name change is part of a set of name changes of DTE Energy subsidiaries to conform their names to the "DTE" brand name. No other changes are contained within this request. This request does not involve a transfer of control over or of an interest in the license for Fermi 2.

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

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

   The proposed amendment changes the name of the owner licensee. The proposed amendment is purely administrative in nature. The functions, powers, resources and management of the owner licensee will not change. Detroit Edison, which will be renamed DTE Electric Company, will remain the licensee of the facility. The proposed changes do not adversely affect accident initiators or precursors, and do not alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety functions is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

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

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

   The proposed amendment is purely administrative in nature. The functions of the owner licensee will not change. These changes do not involve any physical alteration of the plant (i.e., no new or different type of equipment will be installed), and installed equipment is not being operated in a new or different manner. Thus, no new failure modes are introduced.

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

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

   The proposed amendment is a name change to reflect the new name of the owner licensee. The proposed amendment is purely administrative in nature. The functions of the owner licensee will not change. Detroit Edison, which will be renamed DTE Electric Company, will remain the licensee of the facility, and its functions will not change. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There are no changes to setpoints at which protective actions are initiated, and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

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

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

Attorney for licensee: Bruce R. Masters, DTE Energy, General Council – Regulatory, 688 WCB, One Energy Plaza, Detroit, MI 48226-1279.

NRC Branch Chief: Robert D. Carlson.

Luminant Generation Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Nuclear Power Plant, Units 1 and 2, Somervell County, Texas

Date of amendment request: December 19, 2012.

Brief description of amendments: The amendments would revise Technical Specification (TS) 3.8.1, "AC [Alternating Current] Sources - Operating," to revise the Completion Time (CT) for Required Action A.3, "Restore required offsite circuit to OPERABLE status," on one-time basis from 72 hours to 14 days for Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2. The CT extension from 72 hours to 14 days will be used twice while completing the plant modification to install alternate startup transformer (ST) XST1A and will expire on March 31, 2014. After completion of this modification, if ST XST1 should require maintenance or if failure occurs, the alternate ST XST1A can be aligned to the Class 1E buses well within the current CT of 72 hours. Installation of alternate ST will result in improved plant design and will improve the long-term reliability of the 138 kiloVolt (kV) offsite circuit ST.

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

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

   Response: No.

The proposed change will revise the CT for the loss of one offsite source from 72 hours to 14 days to allow two, one-time, 14-day CTs. The proposed two, one-time extensions of the CT for the loss of one offsite power circuit does not significantly increase the probability of an accident previously evaluated. The TS will continue to require equipment that will power safety related equipment necessary to perform any required safety function. The two, one-time extensions of the CT to 14 days does not affect the design of the STs, the interface of the STs with other plant systems, the operating characteristic of the STs, or the reliability of the STs.

The consequence of a LOOP [loss-of-offsite power] event has been evaluated in the CPNPP Final Safety Analysis Report (Reference 8.1 [of application dated December 19, 2012]) and the Station Blackout evaluation. Increasing the CT for one offsite power source twice on a one-time basis from 72 hours to 14 days does not increase the consequences of a LOOP event nor change the evaluation of LOOP events.

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

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

   Response: No.

   The proposed change does not result in a change in the manner in which the electrical distribution subsystems provide plant protection. The proposed change will only affect the time allowed to restore the operability of the offsite power source through a ST. The proposed change does not affect the configuration, or operation of the plant. The proposed change to the CT will facilitate installation of a plant modification which will improve plant design and will eliminate the necessity to shut down both Units if XST1 fails or requires maintenance that goes beyond the current TS CT of 72 hours. This change will improve the long-term reliability of the 138kV offsite circuit ST which is common to both CPNPP Units.

   There are no changes to the STs or the supporting systems operating characteristics or conditions. The change to the CT does not change any existing accident scenarios, nor create any new or different accident scenarios. In addition, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter any of the assumptions made in the safety analysis.

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

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

   Response: No.

   The proposed change does not affect the acceptance criteria for any analyzed event nor is there a change to any safety limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Neither the safety analyses nor the safety analysis acceptance criteria are affected by this change. The proposed change will not result in plant operation in a configuration outside the current design basis. The proposed activity only increases, for two, one-time pre-planned occurrences, the period when the plant may operate with one offsite power source. The margin of safety is maintained by maintaining the ability to safely shut down the plant and remove residual heat.

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

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

<u>Attorney for licensee</u>: Timothy P. Matthews, Esq., Morgan, Lewis and Bockius, 1111 Pennsylvania Avenue, NW, Washington, DC 20004.

<u>NRC Branch Chief</u>: Michael T. Markley.

<u>Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine.</u>

<u>Date of amendment request</u>: January 3, 2012.

<u>Description of amendment request</u>: The amendment proposes to revise License Condition 2.B(6)(d) "Physical Protection." It is proposed to update the title of the Physical Security Plan, from the "Maine Yankee Nuclear Power Station Physical Security Plan", the "Maine Yankee

Nuclear Atomic Power Station Guard Training and Qualification Plan", and the "Maine Yankee Nuclear Power Safeguards Contingency Plan" to the "Maine Yankee Independent Spent Fuel Storage Installation Physical Security Plan."

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

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

   Response: No.

   The proposed amendment is a title change only. There is no reduction in commitments in the Maine Yankee Independent Spent Fuel Storage Installation Physical Security Plan therefore; the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed amendment is a title change only. There is no reduction in commitments in the Maine Yankee Independent Spent Fuel Storage Installation Physical Security Plan 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 amendment is a title change only. There is no reduction in commitments in the Maine Yankee Independent Spent Fuel Storage Installation Physical Security Plan 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: Joseph Fay, Maine Yankee Atomic Power Company, 362 Injun Hollow Road, East Hampton, Connecticut, 06424-3099.

NRC Branch Chief: Michele M. Sampson.

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

Date of amendment request: December 6, 2012.

Description of amendment request: The amendment proposes to revise the Monticello Nuclear Generating Plant (MNGP) Technical Specification (TS) Limiting Condition for Operation 3.10.1, “Inservice Leak and Hydrostatic Testing Operation,” and the associated Bases, to expand its scope to include provisions for temperature excursions greater than 212 °F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in MODE 4. The change is consistent with NRC-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-484, Revision 0, “Use of TS 3.10.1 for Scram Time Testing Activities.”

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee 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.

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

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

Response: No.

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

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

Response: No.

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

The NRC staff has reviewed the 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: Robert D. Carlson.

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

Date of amendment request: December 21, 2012.

Description of amendment request: The amendment proposes to revise the Monticello Nuclear Generating Plant (MNGP) Emergency Plan by revising the Emergency Action Level (EAL) setpoint for the Turbine Building Normal Waste Sump (TBNWS) Monitor. The proposed change reduces the classification of a liquid effluent release via the TBNWS pathway to approximately 48 times the Offsite Does Calculation Manual (ODCM) limit from the current 200 times the ODCM limit, thus establishing a value within the indication capability of the radiation monitor.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee 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 to the emergency plan does not impact the physical function of plant structures, systems, or components (SSCs) or the manner in which SSCs perform their design function. The proposed change neither adversely affects accident initiators or precursors, nor alters design assumptions. The proposed change does not alter or prevent the ability of operable SSCs to perform their intended function to mitigate the consequences of an initiating event within assumed acceptance limits. No operating procedures or administrative controls that function to prevent or mitigate accidents are affected by the proposed change.

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

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

Response: No.

The proposed change does not impact the accident analysis. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), a change in the method of plant operation, or new operator actions. The proposed change will not introduce failure modes that could result in a new accident, and the change does not alter assumptions made in the safety analysis. The proposed change revises an emergency action level (EAL), which establishes the threshold for placing the plant in an emergency classification. EALs are not initiators of any accidents.

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

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation does to the public. The proposed change is associated with the EALs and does not impact operation of the plant or its response to transients or accidents. The change does not affect the technical specifications or the operating license. The proposed change does not involve a change in the method of plant operation, and no accident analyses will be affected by the proposed change. Additionally, the proposed change will not relax

> any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis. The proposed change does not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition.
>
> The revised EAL provides more appropriate and accurate criteria for determining protective measures that should be considered within and outside the site boundary to protect public health and safety. The emergency plan will continue to activate an emergency response commensurate with the extent of degradation of plant safety.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Robert D. Carlson.

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

Date of amendment request: January 4, 2013.

Description of amendment request: The licensee proposed to revise the MNGP Technical Specifications (TS) 3.6.4.3, “Standby Gas Treatment (SGT) System,” TS 3.7.4, “Control Room Emergency Filtration (CREF) System,” and TS 5.5.6, “Ventilation Filter Testing Program (VFTP).” The licensee proposed to modify the TS requirements to operate ventilation systems with charcoal filters from 10 hours each month to 15 minutes in accordance with Technical

Specifications Task Force (TSTF) Traveler TSTF-522, Revision 0, “Revise Ventilation System Surveillance Requirements to Operate for 10 hours per Month.”

Specifically, the licensee proposed to revise the surveillance requirements STET which currently require testing of SGT and CREF Systems, with heaters operating, for a continuous 10 hour period every 31 days without the heaters operating. The associated SRs are proposed to be revised to require operation of these systems for 15 continuous minutes every 31 days. Additionally, the licensee proposed to remove Specification 5.5.6, Item e, under the VFTP, concerning operation of the SGT and CREF Systems heaters.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration, which is provided 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 replaces existing SRs to operate the SGT System and CREF System equipped with electric heaters for a continuous 10 hour period every 31 days with a requirement to operate the systems for 15 continuous minutes (without the heaters operating) and removes a no longer required SR under the VFTP.

   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 changes do not involve a significant increase in the consequences of an accident.

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

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

Response: No.

The proposed changes replaces existing SRs to operate the SGT System and CREF System equipped with electric heaters for a continuous 10 hour period every 31 days with a requirement to operate the systems for 15 continuous minutes (without the heaters operating) and removes a no longer required SR under the VFTP.

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

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

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

Response: No.

The proposed changes replaces existing SRs to operate the SGT System and CREF System equipped with electric heaters for a continuous 10 hour period every 31 days with a requirement to operate the systems for 15 continuous minutes (without the heaters operating) and removes a no longer required SR under the VFTP. Testing requirements will be revised and will continue to demonstrate that the LCOs are met and the system components are capable of performing their intended safety functions.

The proposed changes are consistent with regulatory guidance. Therefore, it is concluded that these changes do not involve a significant reduction in a margin of safety.

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

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

<u>NRC Branch Chief</u>: Robert D. Carlson.

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: December 13, 2012.

Description of amendment request: The proposed amendments would revise the Prairie Island Nuclear Generating Plant Emergency Plan by revising certain emergency action levels described in the plan.

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 Emergency Plan emergency action levels for classification of liquid effluent releases and determining fuel clad barrier loss. These changes propose to use installed plant radiation monitors differently but do not involve any physical plant changes.

   The Emergency Plan emergency action levels and installed plant radiation monitors are not accident initiators and therefore the proposed changes do not involve an increase in the probability of an accident. The proposed emergency action level changes do not affect the capability of any structures, system or components to mitigate a design basis accident. Thus the proposed changes do not involve a significant increase in the consequences of an accident.

   Therefore, the proposed Emergency Plan emergency action level 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 Emergency Plan emergency action levels for classification of liquid effluent releases and determining fuel clad barrier loss. These changes propose to use installed plant radiation monitors differently but do not involve any physical plant changes.

The proposed Emergency Plan emergency action level changes 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 ensures that the plant Emergency Plan is effectively and consistently implemented. 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 Emergency Plan emergency action level 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 Emergency Plan emergency action levels for classification of liquid effluent releases and determining fuel clad barrier loss. These changes propose to use installed plant radiation monitors differently but do not involve any physical plant changes.

Margin of safety is provided by the ability of accident mitigation structures systems or components to perform at their analyzed capability. The changes proposed in this license amendment request do not affect the capability of any equipment to perform its accident mitigation function. Thus, no margin of safety is reduced as part of this change.

Therefore, the proposed Emergency Plan emergency action level 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: Robert D. Carlson.

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

Date of amendment request: February 7, 2013.

Description of amendment request: The proposed change would amend Combined License Nos.: NPF-93 and NPF-94 for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 in regard to the Primary Sampling System (PSS) by: (1) replacing containment air return check valve PSS-PL-V024 with a solenoid-operated valve, and (2) redesigning the PSS inside-containment header and adding a PSS containment penetration.

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

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

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

   Response: No

   The Primary Sampling System (PSS) provides the safety-related function of preserving containment integrity by isolation of the PSS lines penetrating containment. The proposed amendment will enhance the ability of the PSS to perform its nonsafety-related function of providing the capability to obtain reactor coolant and containment atmosphere samples, while maintaining the ability of the PSS to perform its safety-related containment isolation function. The replacement of a check valve with a solenoid-operated containment isolation valve and the redesigned inside-containment header does not affect the safety-related function of isolating the PSS lines for containment isolation. The components added by this proposed activity, including tubing and the solenoid-operated containment isolation valve, are designed to the same codes and standards as other components addressed in the certified design that perform similar functions. The additional PSS containment penetration is a passive extension of containment and is identical in form, fit, and function to other PSS sampling containment penetrations currently addressed in the certified AP1000 plant design. The addition of a new PSS containment penetration will not change the maximum allowable leakage rate allowed by Technical Specifications and verified periodically in accordance with regulations. Furthermore, the proposed PSS configuration changes will neither impact any accident source term parameter or fission product barrier nor affect radiological dose consequence analysis.

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 additional containment penetration is similar in form, fit, and function to the PSS penetrations that are currently described in the Updated Final Safety Analysis Report. Because the PSS changes use valve types, piping, and a containment penetration consistent with those already described in the Updated Final Safety Analysis Report, no new failure modes or equipment failure initiators are introduced by these changes. Accordingly, the proposed changes do not create any new malfunctions, failure mechanisms, or accident initiators.

   Therefore, the proposed amendment will 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 containment isolation function is not changed by this activity and is bounded by the existing design. The proposed PSS containment penetration is similar in form, fit, and function to other containment penetrations in similar applications in the current certified AP1000 plant design. The additional PSS containment penetration is an extension of containment, and, therefore, does not affect containment or its ability to perform its design function. The addition of PSS components, including the solenoid-operated containment isolation valve, the additional PSS containment penetration, and the associated tubing, do not exceed or alter a design basis or safety limit. Because the containment isolation function, containment leakage rate limit, potential containment leakage, and protective shielding are not changed by this activity and are bounded by the existing design, there is no change to any current 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: Ms. Kathryn M. Sutton, Morgan, Lewis & Bockius LLC, 1111 Pennsylvania Avenue, NW, Washington, DC 20004-2514.

NRC Branch Chief: Lawrence Burkhart, Acting.

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

Date of amendment request: February 14, 2013.

Description of amendment request: The proposed change would amend Combined License Nos.: NPF-93 and NPF-94 for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 in regard to the structural module stud size and spacing by increasing the carbon steel vertical stud spacing, decreasing the stainless steel stud diameter, and decreasing the stainless steel vertical and horizontal stud spacing in accordance with the design basis.

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

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

   Response: No

   The design function of the containment modules is to support the reactor coolant system components and related piping systems and equipment. The design functions of the affected structural module in the auxiliary building are to provide support and protection for new and spent fuel and the equipment needed to support fuel handling, cooling, and storage in the spent fuel racks, and to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located outside the containment building. The design function of the shear studs it to transfer loads into the concrete of the structural modules. The proposed change corrects a drawing note regarding shear stud size and spacing for structural wall modules to be consistent with the underlying design basis calculations, which are more conservative. The thickness, geometry, and strength of the structures are not adversely altered. The properties of the concrete included in the modules are not altered. As a result, the design function of the structural modules is not adversely affected by the proposed change. There is no change to plant, systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to normal operation or postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor does the change described create any new accident precursors.

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

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

   Response: No

   The proposed change corrects a drawing note regarding shear stud size and spacing for structural wall modules to be consistent with the underlying design basis calculations. Stud spacing and sizing are updated such that stud loadings are within acceptable limits and that the structural module acts in a composite manner. The thickness, geometry, and strength of the structures are not adversely altered. The properties of the concrete included in the modules are not altered. The change to the internal design of the structural modules does not create any new accident precursors. As a result, the design function of the modules is not adversely affected by the proposed change.

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

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

Response: No

The criteria and requirements of AISC-N690 provide a margin of safety to structural failure. The design of the shear studs for the structural wall modules conforms to criteria and requirements in AISC-N690 and therefore maintains the margin of safety. The proposed change corrects a drawing note regarding shear stud size and spacing for the structural wall modules so as to be consistent with the underlying design basis calculations. There was no change to the method of evaluation from that used in the design basis calculations.

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

NRC Branch Chief: Lawrence Burkhart, Acting.

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

Date of amendment request: February 7, 2013 and revised on February 14, 2013.

Description of amendment request: The proposed change would amend Combined License Nos.: NPF-93 and NPF-94 for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 to allow the use of concentrically and eccentrically braced frames in the turbine building main area and modify the applicable design code.

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 Tier1 in accordance with 52.63(b)(1).

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

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

   Response: No

   The turbine building bracing design is changed to a mixed bracing system which uses special concentric and eccentric bracing. The turbine building does not contain safety-related systems or components. The main area of the turbine building continues to meet its design function of preventing a turbine building collapse from impairing the integrity of seismic Category I structures, systems, or components. The first bay of the turbine building is designed to prevent the collapse of the main area of the Turbine Building onto the Nuclear Island during a seismic event. The proposed changes do not affect or impact this design capability. Therefore, the response of the safety related systems, structures, and components in the Nuclear Island to earthquakes and postulated accidents are not affected by the bracing of the turbine building. Based on the above, there is no change in the probability of an accident previously evaluated. The activity does not introduce a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that result in significant fuel cladding failures. Accordingly, there is no change 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 turbine building bracing design is changed to a mixed bracing system which uses Special Concentrically Braced Framing (SCBF) and Eccentrically Braced Framing (EBF). The main area of the turbine building continues to meet its design function of preventing a turbine building collapse from impairing the integrity of seismic Category I structures, systems, or components. The design function of the turbine building first bay to provide the intended limitations to a potential collapse onto the nuclear island during a seismic event is retained. The turbine building structure does not involve any accident initiating component and therefore, changes to use SCBF and EBF would not introduce new accident components or faults.

   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

   Use of a mixed bracing system and changing the structural code design for the turbine building main area continue to meet the design function of preventing a turbine building collapse from impairing the integrity of seismic Category I Structures, Systems, and Components. In addition, the first bay of the turbine building continues to be designed to seismic Category II requirements to prevent a turbine building collapse from impairing the integrity of the seismic Category I nuclear island structures, systems and components. This portion of the turbine building and its design is unchanged by the proposed amendment. Maintaining the seismic Category II rating for the turbine building first bay, along with continuing to meet the design function for the non-safety , non-seismic design of the turbine building main area preserves the current structural safety margins.

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

NRC Branch Chief: Lawrence Burkhart, Acting.

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: January 11, 2013.

Description of amendment request: The proposed change would amend Combined License Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 in regard to the Chemical and Volume Control System (CVS) by: (1) providing a spring-assisted check valve around the air-operated Reactor coolant System (RCS) Purification Return Line Stop Check Valve , (2) replacing the CVS zinc addition inboard containment isolation lift check valve with an air-operated globe valve and a thermal relief valve and (3) separating the zinc and hydrogen injection paths and relocate the zinc injection path.

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

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

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

   Response: No

   The changes to provide a spring-assisted check valve located in the bypass line around the makeup stop check valve would continue to meet the existing design functions because the ASME Boiler and Pressure Vessel Code (ASME Code) Section III valves will maintain the flow isolation design function and preserve the Reactor Coolant System (RCS) pressure boundary safety function. The replacement of the Chemical and Volume Control System (CVS) zinc addition inboard containment isolation lift check valve with an air operated globe valve and addition of a pressure relief valve would continue to meet the containment isolation and RCS pressure boundary design functions because the replacement valves will be designed, analyzed, tested and qualified, including seismic qualification, to ASME Code Section III requirements. Separating the zinc and hydrogen injection paths and relocating the zinc injection point would continue to meet containment boundary requirements, including containment isolation and in-service testing, and preserve the RCS pressure boundary safety functions because the revised containment isolation configuration is consistent with those described in 10 CFR 50, Appendix A, General Design Criterion (GDC) 55, and the additional valves and piping will be qualified to ASME Code Section III. Because the proposed CVS changes would preserve the CVS safety-related design functions, the probability of an accident previously evaluated is not affected.

   The CVS safety functions have been preserved, because the proposed CVS configuration changes, including revised valve types, will perform the same safety functions as the current design. The proposed CVS configuration changes would neither impact any accident source term parameter or fission product barrier nor affect radiological dose consequence analysis.

   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 additional containment penetration is similar in form, fit, and function to the CVS combined zinc/hydrogen containment penetration that is currently described in the Updated Final Safety Analysis Report. Because the CVS changes use valve types, piping, and a containment penetration consistent with those already described in the Updated Final Safety Analysis Report, no new failure modes or equipment failure initiators are introduced by these changes. Accordingly, the proposed changes do not create any new malfunctions, failure mechanisms, or accident initiators.

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

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

Response: No

The containment isolation and pressure relief functions would not be changed by this activity and are consistent with the existing design. The proposed CVS containment penetration is similar in form, fit, and function to existing CVS combined zinc/hydrogen containment penetration and, therefore, does not affect containment or its ability to perform its design function. The addition of these CVS components, including piping, a spring-assisted check valve, an air-operated containment isolation valve, a thermal relief valve and the additional CVS containment penetration do not impact a design basis or safety limit. Because the CVS design functions of controlling the RCS oxygen concentration, reducing radiation fields, containment isolation and overpressure protection within existing limits are not changed by this activity and are bounded by the existing design, there is no change to any current margin of safety.

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

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

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

NRC Branch Chief: Lawrence Burkhart, Acting.

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: February 7, 2013 and revised on February 15, 2013.

Description of amendment request: The proposed change would amend Combined License Nos.: NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 to allow the use of concentrically and eccentrically braced frames in the turbine building main area and modify the applicable design code.

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 Tier1 in accordance with 52.63(b)(1).

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

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

   Response: No

   The turbine building bracing design is changed to a mixed bracing system which uses special concentric and eccentric bracing. The turbine building does not contain safety-related systems or components. The main area of the turbine building continues to meet its design function of preventing a turbine building collapse from impairing the integrity of seismic Category I structures, systems, or components. The first bay of the turbine building is designed to prevent the collapse of the main area of the Turbine Building onto the Nuclear Island during a seismic event. The proposed changes do not affect or impact this design capability. Therefore, the response of the safety related systems, structures, and components in the Nuclear Island to earthquakes and postulated accidents are not affected by the bracing of the turbine building. Based on the above, there is no change in the probability of an accident previously evaluated. The activity does not introduce a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that result in significant fuel cladding failures. Accordingly, there is no change 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 turbine building bracing design is changed to a mixed bracing system which uses Special Concentrically Braced Framing (SCBF) and Eccentrically Braced Framing (EBF). The main area of the turbine building continues to meet its design function of preventing a turbine building collapse from impairing the integrity of seismic Category I structures, systems, or components. The design function of the turbine building first bay to provide the intended limitations to a potential collapse onto the nuclear island during a seismic event is retained. The turbine building structure does not involve any accident initiating component and therefore, changes to use SCBF and EBF would not introduce new accident components or faults.

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

Use of a mixed bracing system and changing the structural code design for the turbine building main area continue to meet the design function of preventing a turbine building collapse from impairing the integrity of seismic Category I Structures, Systems, and Components. In addition, the first bay of the turbine building continues to be designed to seismic Category II requirements to prevent a turbine building collapse from impairing the integrity of the seismic Category I nuclear island structures, systems and components. This portion of the turbine building and its design is unchanged by the proposed amendment. Maintaining the seismic Category II rating for the turbine building first bay, along with continuing to meet the design function for the non-safety, non-seismic design of the turbine building main area preserves the current structural safety margins.

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

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

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

NRC Branch Chief: Lawrence Burkhart, Acting.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: December 13, 2012.

Description of amendment request: The amendment would revise Technical Specification (TS) 3.7.9, "Ultimate Heat Sink (UHS)," to incorporate more restrictive UHS level and pond temperature limits which are specified in Surveillance Requirements (SRs) 3.7.9.1 and 3.7.9.2, respectively. In addition, new SR 3.7.9.4 would be added to verify that the UHS cooling tower fans respond appropriately to automatic start signals.

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.

   There are no design changes associated with the proposed amendment. All design, material, and construction standards that were applicable prior to this amendment request will continue to be applicable. The proposed change will not adversely affect accident initiators or precursors or adversely alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained with respect to such initiators or precursors. The proposed changes do not affect the way in which safety-related systems perform their functions.

All accident analysis acceptance criteria will continue to be met with the proposed changes. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [final safety analysis report]. The applicable radiological dose acceptance criteria will continue to be met.

The intent of the modified UHS water level and temperature limits for TS 3.7.9, as proposed, is to ensure that the UHS can perform its specified safety function for accident mitigation, including consideration of its 30-day mission time. The proposed surveillance limits are more restrictive and are based on an analysis that includes credit given to specific operator actions (with assumed completion times) not previously assumed. However, the operator actions are reasonable and have been established in accordance with NRC-approved guidance. Further, they have been simulator verified and proven to be capable of being met by plant operators under applicable accident scenarios.

The crediting of these operator actions is consistent with the plant's current licensing basis which already credits operator action to provide long-term protection of the UHS following an accident. These actions, in conjunction with the more restrictive proposed UHS water temperature and level surveillance limits, support the plant's existing accident analysis such that there is no change in analyzed consequences. In light of these considerations, there is no significant increase in the consequences of any accident previously evaluated with regard to the assumed operator actions and revised UHS water level and temperature limits, as proposed. The proposed change adds additional controls to the Technical Specifications but does not physically alter safety-related systems or affect the way in which safety-related systems perform their functions per the intended plant design.

As such, the proposed change will not alter or prevent the capability of structures, systems, and components (SSCs) to perform their intended functions for mitigating the consequences of an accident and meeting applicable acceptance limits. 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.

   With respect to any new or different kind of accident, there are no proposed design changes nor are there any changes in the method by which any safety-related plant SSC performs its specified safety function. The proposed change will not affect the normal method of plant operation. No new transient precursors will be introduced as a result of this amendment. The reanalysis discussed herein addresses new large break LOCA [loss-of-coolant accident] scenarios with assumptions, including single failures, aimed at maximizing the UHS temperature and minimizing the UHS inventory.

   The proposed change adds requirements to the Technical Specifications. The change does not involve a physical modification of the plant. The UHS level and temperature limits within which the plant is normally operated are being changed in the conservative direction. Appropriate changes have been made to the emergency operating procedures relied upon to mitigate a design basis event. The change does not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. The changes to the ultimate heat sink (UHS) surveillance limits are in the conservative direction.

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

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

   Response: No.

   There will be no effect on those plant systems necessary to assure the accomplishment of protection functions associated with reactor operation or the reactor coolant system. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (FΔH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other limit and associated margin of safety. Required shutdown margins in the COLR [core operating limits report] will not be changed.

   The proposed change does not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. The proposed change would add Technical Specification Surveillance Requirements for assuring the automatic closure of the UHS cooling tower bypass valves when required and the automatic start of the UHS

> cooling tower fans and their transition from slow speed to fast speed when required. The extent of Callaway's conformance to NRC Regulatory Guide (RG) 1.27 is discussed in FSAR Site Addendum Table 9.2-5 (see Attachment 4 to this Enclosure [to the submittal]). RG 1.27 requires that the UHS be sized for 30 day post-LOCA operation; however, it does not specify a margin value above that 30-day requirement. During initial plant licensing (Callaway Safety Evaluation Report, NUREG-0830, Supplement 4, Section 2.4.4) a UHS level margin of 50% was accepted in lieu of a more restrictive minimum Technical Specification water level of 834 feet mean sea level (16 feet above the reference pond bottom) and a thermal and hydrologic analysis of the ESW [essential service water] and UHS. In this amendment request SR 3.7.9.1 is being changed to adopt the former and the supporting EF-123 analysis addresses the latter. The SER [safety evaluation report] Supplement 4 discussion, copied in Section 2.2 of this Evaluation, will no longer be applicable upon NRC approval of this license amendment request.
>
> As such, the proposed change does not involve a significant reduction in a margin of safety as defined in any regulatory requirement or guidance document.

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 O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Michael T. Markley.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: December 20, 2012.

Description of amendment request: The amendment would revise a methodology in the licensing basis as described in the Final Safety Analysis Report - Standard Plant to include damping values for the seismic design and analysis of the integrated head assembly that are consistent with the recommendations of NRC Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Revision 1, March 2007.

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

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

   Response: No.

   The proposed change would allow use of critical damping values consistent with the recommendations of RG [Regulatory Guide] 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Revision 1, dated March 2007, for the seismic design and analysis of the IHA [integrated head assembly].

   The RG 1.61, Revision 1, Table 1 note allowing use of a “weighted average” for design-basis SSE [safe shutdown earthquake] damping values applicable to steel structures of different connection types, is also applied to determine the IHA design-basis OBE [operating basis earthquake] damping values. RG 1.61, Revision 1, Table 2 for OBE damping values does not contain the same note found in Table 1. However use of the note for the determination of the OBE damping value is consistent with the use of the note for the determination of the SSE damping values, and a weighted average more realistically represents the IHA structure. RG 1.61, Revision 1, specifies the damping values that the NRC staff currently considers acceptable for complying with the agency's regulations and guidance for seismic analysis. Revision 1 incorporates the latest data and information, and reduces unnecessary conservatism in specification of damping values for seismic design and analysis of SSCs [structures, systems, and components].

   The proposed change does not change the design functions of the IHA or its response to design-basis events, nor does it affect the capability of related SSCs to perform their design or safety functions. The use of the proposed damping values in the seismic design and analysis of the IHA is related to the ability of the IHA to function in response to design-basis seismic events, and is unrelated to the probability of occurrence of those events, or other previously evaluated accidents. Therefore, the proposed change will not have any impact on the probability of an accident previously evaluated.

The proposed damping values are an element of the seismic analyses performed to confirm the ability of the IHA to function under postulated seismic events while maintaining resulting stresses within ASME [American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*] Section III allowable values. Therefore, the use of damping values consistent with the recommendations of RG 1.61, Revision 1 does not result in an increase in the consequences of accidents previously evaluated.

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

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

   Response: No.

   The proposed change does not involve changes to any plant SSCs, nor does it involve changes to any plant operating practice or procedure. The damping values are an element of the seismic analyses performed to confirm the ability of the IHA to function under postulated seismic events while maintaining resulting stresses within ASME Section III allowable values. Therefore, no credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases are created that would create the possibility of a new or different kind of accident.

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

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

   Response: No.

   The design basis of the plant requires structures to be capable of withstanding normal and accident loads including those from a design basis earthquake. The proposed change would allow the use of damping values in the IHA seismic analyses that are, in general, more realistic and, thus, more accurate than the damping values recommended in RG 1.61, Revision 0, used in the original analysis for the SSE, or the plant specific damping values used in the original analysis for the OBE. The damping values in RG 1.61, Revision 0, were based on limited data, expert opinion, and other information available in 1973. NRC and industry research since 1973 shows that the damping values provided in the original version of RG 1.61 may not reflect realistic damping values for SSCs. RG 1.61, Revision 1, therefore, provides damping values based on the updated research results that predict and estimate damping

> values for seismic design of SSCs in nuclear power plants, and similarly should not be regarded as an arbitrary lowering of the margins 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 O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Michael T. Markley.

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

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

A notice of consideration of issuance of amendment to facility operating license or combined license, as applicable, proposed no significant hazards consideration determination, and opportunity for a hearing in connection with these actions, was published in the *Federal Register* as indicated.

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through the Agencywide Documents Access and Management System (ADAMS) in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR's Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr.resource@nrc.gov.

Carolina Power and Light Company, et al., Docket No. 50-261, H.B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: March 16, 2012, as supplemented by letter dated August 16, 2012.

Brief Description of amendment: The amendment revised the Technical Specifications (TSs) to make corrections in TS Table 3.3.1-1 for Overtemperature Delta Temperature consistent with NUREG-1431, Revision 3, "Standard Technical Specifications Westinghouse Plants."

Date of issuance: February 13, 2013.

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

Amendment No.: 231.

Renewed Facility Operating License No. DPR-23: Amendment changed the license and TSs.

Date of initial notice in *Federal Register*: April 17, 2012 (77 FR 22811). The supplement. dated August 16, 2012, 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 February 13, 2013.

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 application for amendment: June 6, 2012, as supplemented by letter dated. November 19, 2012.

Brief description of amendment: The proposed amendment modifies Braidwood and Byron technical specifications (TS) to add a Note to surveillance requirements (SRs) 3.3.1.7, 3.3.1.8, and 3.3.1.12 in TS 3.3.1, “Reactor Trip System (RTS) Instrumentation,” and SRs 3.3.2.2 and 3.3.2.6 in TS 3.3.2, “Engineered Safety Features Actuation System (ESFAS) Instrumentation,” to exclude the Solid State Protection System input relays from the Channel Operational Test Surveillance for RTS and ESFAS functions with installed bypass capability which the U.S. Nuclear Regulatory Commission (NRC) approved by letters dated March 30, and April 9, 2012.

Date of issuance: February 6, 2013.

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

Amendment Nos.: 171 for Braidwood Station, Units 1 and 2, and 178 for Byron Station, Unit Nos. 1 and 2, respectively.

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: September 4, 2012 (77 FR 53927).

The November 19, 2012, supplement contained clarifying information and did not change the NRC staff’s initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 6, 2013.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50-260 and 50-296, Browns Ferry Nuclear Plant (BFN), Units 2 and 3, Limestone County, Alabama

Date of application for amendments: February 25, 2011, as supplemented by letters dated September 15, 2011, July 30, 2012, and January 24, 2013. The enclosure to the July 30, 2012, letter superseded, in its entirety, the enclosure to the February 25, 2011, letter.

Brief description of amendments: The amendments delete the BFN, Units 2 and 3, Technical Specification (TS) Surveillance Requirement 3.5.1.12, which requires the verification of the capability to automatically transfer the power supply from the normal source to the alternate source for each Low-Pressure Coolant Injection subsystem inboard injection valve and each recirculation pump discharge valve on a 24-month frequency. In addition, these amendments approve the use of a modified loss-of-coolant accident (LOCA) methodology that requires revising TS 5.6.5.b to include a reference to the modified LOCA methodology. Also, the amendments revise TSs 3.3.1.1, 5.6.5.a, and 5.6.5.b to include the modified LOCA methodology and the oscilliation power range monitor upscale function period based detection algorithm setpoint limits.

Date of issuance: February 15, 2013.

Effective date: The amendments are effective as of this date of issuance. For Unit 2, the amendment shall be implemented prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2013 refueling outage. For Unit 3, changes to TSs 5.6.5 and 3.3.1 shall be implemented within 60 days of issuance. The remaining changes shall be implemented prior to entering Mode 3 from the spring 2014 refueling outage.

Amendment Nos.: Unit 1 - 309 and Unit 2 - 268.

Renewed Facility Operating License Nos. DPR-52 and DPR-68: Amendments revised the licenses and TSs.

Date of initial notice in *Federal Register:* The original application dated February 25, 2011, was noticed on May 3, 2011 (76 FR 24930). The supplement dated July 30, 2012, was noticed on November 5, 2012 (77 FR 66490). The supplement dated January 24, 2013, provided additional information that clarified the licensee’s July 30, 2012, submittal, did not expand the scope of the application as noticed and did not change the NRC staff's proposed no significant hazards consideration determination as published in the FR on November 5, 2012 (77 FR 66490).

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

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50-339, North Anna Power Station, Unit No. 2, Louisa County, Virginia

Date of application for amendment: May 11, 2012.

Brief Description of amendment: The amendment would revise the Technical Specification (TS) 3.1.7, “Rod Position Indication” to allow two demand position indicators in one or more banks to be inoperable for up to 4 hours. This change is proposed as a temporary change to the TS for the current operating cycle and is proposed as a footnote to the current TS Limiting Condition for Operation (LCO) Section 3.1.7, Condition D.

Date of issuance: February 14, 2013.

Effective date: As of the date of issuance and shall be implemented within the end of operating Cycle 22.

Amendment No.: 251.

Renewed Facility Operating License No. NPF-7: Amendment changes the license and the TS.

Date of initial notice in *FEDERAL REGISTER*: June 12, 2012 (77 FR 35077).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 2013.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 25th day of February 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Louise Lund, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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