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[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2018-0266]

Biweekly Notice

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from October 23, 2018, to November 5, 2018. The last biweekly notice was published on November 6, 2018.

DATES: Comments must be filed by [INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]. A request for a hearing must be filed by [INSERT DATE 60 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*].

ADDRESSES: You may submit comments by any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2018-0266**. Address questions about Docket IDs in Regulations.gov to Jennifer Borges; telephone: 301-287-9127; e-mail: Jennifer.Borges@nrc.gov. For technical questions, contact the individual listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **Mail comments to:** May Ma, Office of Administration, Mail Stop: TWFN-7-A60M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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

FOR FURTHER INFORMATION CONTACT: Janet Burkhardt, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; telephone: 301-415-1384, e-mail: Janet.Burkhardt@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID **NRC-2018-0266**, facility name, unit number(s), plant docket number, application date, and subject when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2018-0266**.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public

Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select “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. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document.

- **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-2018-0266**, facility name, unit number(s), plant docket number, application date, and subject in your comment submission.

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

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

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

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

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

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period if 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. If the Commission takes 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. If the

Commission makes a final no significant hazards consideration determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

Within 60 days after the date of publication of this notice, any persons (petitioner) whose interest may be affected by this action may file a request for a hearing and petition for leave to intervene (petition) with respect to the action. Petitions shall be filed in accordance with the Commission's "Agency Rules of Practice and Procedure" in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309. The NRC's regulations are accessible electronically from the NRC Library on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Alternatively, a copy of the regulations is available at the NRC's Public Document Room, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. If a petition is filed, the Commission or a presiding officer will rule on the petition and, if appropriate, a notice of a hearing will be issued.

As required by 10 CFR 2.309(d) the petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements for standing: (1) the name, address, and telephone number of the petitioner; (2) the nature of the petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the 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 petitioner's interest.

In accordance with 10 CFR 2.309(f), the petition must also set forth the specific contentions which the petitioner seeks to have litigated in the proceeding. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner must 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 petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue. The petition must include sufficient information to show that a genuine dispute exists with the applicant or licensee on a material issue of law or fact. Contentions must be limited to matters within the scope of the proceeding. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to satisfy the requirements at 10 CFR 2.309(f) 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. Parties have the opportunity to participate fully in the conduct of the hearing with respect to resolution of that party's admitted contentions, including the opportunity to present evidence, consistent with the NRC's regulations, policies, and procedures.

Petitions must be filed no later than 60 days from the date of publication of this notice. Petitions and motions for leave to file new or amended contentions that are filed after the deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i) through (iii). The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to establish 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 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 the amendment unless the Commission finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR part 2.

A State, local governmental body, Federally-recognized Indian Tribe, or agency thereof, may submit a petition to the Commission to participate as a party under 10 CFR 2.309(h)(1). The petition should state the nature and extent of the petitioner's interest in the proceeding. The petition should be submitted to the Commission no later than 60 days from the date of publication of this notice. The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document, and should meet the requirements for petitions set forth in this section, except that under 10 CFR 2.309(h)(2) a State, local governmental body, or Federally-recognized Indian Tribe, or agency thereof does not need to address the standing requirements in 10 CFR 2.309(d) if the facility is located within its boundaries. Alternatively, a State, local governmental body, Federally-recognized Indian Tribe, or agency thereof may participate as a non-party under 10 CFR 2.315(c).

If a hearing is granted, any person who is not a party to the proceeding and is not affiliated with or represented by a party may, at the discretion of the presiding officer, be permitted to make a limited appearance pursuant to the provisions of 10 CFR 2.315(a). A person making a limited appearance may make an oral or written statement of his or her position on the issues but may not otherwise participate in the proceeding. A limited appearance may be made at any session of the hearing or at any prehearing conference, subject to the limits and conditions as may be imposed by the presiding officer. Details regarding the opportunity to make a limited appearance will be provided by the presiding officer if such sessions are scheduled.

B. Electronic Submissions (E-Filing).

All documents filed in NRC adjudicatory proceedings, including a request for hearing and petition for leave to intervene (petition), 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 that request to participate under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007, as amended at 77 FR 46562; August 3, 2012). 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. Detailed guidance on making electronic submissions may be found in the Guidance for Electronic Submissions to the NRC and on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html*. 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 (1) request a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign submissions and access the E-Filing system for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a petition or other adjudicatory document (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

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

A person filing electronically using the NRC's adjudicatory E-Filing system may seek assistance by contacting the NRC's Electronic Filing Help Desk through the "Contact Us" link located on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail to *MSHD.Resource@nrc.gov*, or by a toll-free call at 1-866-672-7640. The NRC Electronic Filing Help Desk is available between 9 a.m. and 6 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 stating why there is good cause for not filing electronically and 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, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff. Participants filing adjudicatory documents 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 *https://adams.nrc.gov/ehd*, unless excluded pursuant to an order of the Commission or the presiding officer. If you do not have an NRC-issued digital ID certificate as described above, click cancel when the link requests certificates and you will be automatically directed to the NRC's electronic hearing dockets where you will be able to access any publicly available documents in a particular hearing docket. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or personal phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. For example, in some instances, individuals provide home addresses in order to demonstrate proximity to a facility or site. 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.

For further details with respect to these license amendment applications, see the application for amendment which is available for public inspection in ADAMS and at the NRC's PDR. For additional direction on accessing information related to this document, see the "Obtaining Information and Submitting Comments" section of this document.

<u>Duke Energy Carolinas, LLC, Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2 (Catawba), York County, South Carolina</u>

Date of amendment request: July 19, 2018. A publicly-available version is in ADAMS under Accession No. ML18200A252.

Description of amendment request: The amendments would modify the Catawba Updated Final Safety Analysis Report (UFSAR), Section 6.2.4.2.2, "Containment Valve Injection Water System [CVIWS]," to remove the CVIWS supply from specified Safety Injection (NI) and Containment Spray (NS) Containment Isolation Valves (CIVs), and to exempt these CIVs from Type C Local Leak Rate Testing (LLRT). Additionally, the amendments would modify UFSAR, Table 6-77, "Containment Isolation Valve Data," to make corresponding changes.

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 amendment request is to remove select Containment Isolation Valves from the Local Leak Rate Test (LLRT) program. These valves were originally included in the LLRT under 10 CFR 50, Appendix J, in what is now Option A. [Catawba] has been approved for 10 CFR 50, Appendix J, Option B under License Amendment No. 192/184. Under Option B, valves may be exempted from LLRT Type C testing if they are not a potential containment atmosphere leakage path. Based on the design and operation of the NI and NS Systems, the valves do not constitute a containment atmospheric leakage path as covered in the Safety Evaluation. Since the valves are not a leakage path, there is no impact on the consequence of an accident. Moreover, the valves are not a part of the Reactor Coolant Pressure Boundary, thus they do not affect the probability of an accident.

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

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

Response: No.

The systems design and operation are not changing. This test exemption does not change the way the valves are used as a part of the NI and NS Systems. A detailed Failure Modes and Effects Analysis was completed to confirm the system operation would meet the containment isolation design function.

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.

The test exemption is within existing regulatory requirements. The application of a closed loop outside of containment is appropriate and consistent with regulatory positions. With containment integrity maintained within the allowable regulatory framework, there is no reduction in the margin of safety.

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

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

Attorney for licensee: Kate B. Nolan, Deputy General Counsel, Duke Energy Carolinas, LLC, 550 South Tryon Street – DEC45A, Charlotte, NC 28202-1802.

NRC Branch Chief: Michael T. Markley.

Exelon FitzPatrick, LLC and Exelon Generation Company, LLC, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant (FitzPatrick), Oswego County, New York

Date of amendment request: October 2, 2018. A publicly-available version is in ADAMS under Accession No. ML18275A060.

Description of amendment request: The amendment would modify the Technical Specifications concerning a change to the method of calculating core reactivity for the purpose of performing the reactivity anomaly surveillance at FitzPatrick.

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

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

   Response: No.

   The proposed Technical Specification change does not affect any plant systems, structures, or components designed for the prevention or mitigation of previously evaluated accidents. The amendment would only change how the reactivity anomaly surveillance is performed. Verifying that the core reactivity is consistent with predicted values ensures that accident and transient safety analyses remain valid. This amendment changes the Technical Specification requirements such that, rather than performing the surveillance by comparing predicted to actual control rod density, the surveillance is performed by a direct comparison of k_{eff}. Present day on-line core monitoring systems, such as the one in use at the James A. FitzPatrick Nuclear Power Plant [(JAFNPP)], Unit 1 are capable of performing the direct measurement of reactivity.

   Therefore, since the reactivity anomaly surveillance will continue to be performed by a viable method, the proposed amendment does not involve a significant increase in the probability or consequence of a previously evaluated accident.

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

Response: No.

This Technical Specifications amendment request does not involve any changes to the operation, testing, or maintenance of any safety-related, or otherwise important to safety systems. All systems important to safety will continue to be operated and maintained within their design bases. The proposed changes to the reactivity anomaly Technical Specifications will only provide a new, more efficient method of detecting an unexpected change in core reactivity.

Since all systems continue to be operated within their design bases, no new failure modes are introduced and the possibility of a new or different kind of accident is not created.

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

Response: No.

This proposed Technical Specifications amendment proposes to change the method for performing the reactivity anomaly surveillance from a comparison of predicted to actual control rod density to a comparison of predicted to actual k_{eff}. The direct comparison of k_{eff} provides a technically superior method of calculating any differences in the expected core reactivity. The reactivity anomaly surveillance will continue to be performed at the same frequency as is currently required by the Technical Specifications, only the method of performing the surveillance will be changed. Consequently, core reactivity assumptions made in safety analyses will continue to be adequately verified.

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: Donald P. Ferraro, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Suite 305, Kennett Square, PA 19348.

NRC Branch Chief: James G. Danna.

Exelon Generation Company (EGC), LLC, Docket No. 50-461, Clinton Power Station (CPS), Unit No. 1, DeWitt County, Illinois

Date of amendment request: September 28, 2018. A publicly-available version is in ADAMS under Accession No. ML18271A217.

Description of amendment request: The amendment would make Technical Specification (TS) changes that are consistent with NRC-approved Industry Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-476, Revision 1. The availability of this TS improvement was announced in the *Federal Register* on May 23, 2007 (72 FR 29004).

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 modifies the TS to allow the use of the improved BPWS [Banked Position Withdrawal Sequence] during shutdowns if the conditions of NEDO-33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004 [ADAMS Accession No. ML042230366], have been satisfied. The justifications to support the specific TS changes are consistent with the approved topical report and TSTF-476, Revision 1. Since the change only involves changes in control rod sequencing, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident after adopting TSTF-476 are no different than the consequences of an accident prior to adopting TSTF-476. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The control rod drop accident (CRDA) is the design basis accident for the subject TS changes. This change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

   Response: No.

   The proposed change, TSTF-476, Revision 1, incorporates the improved BPWS, previously approved in NEDO-33091-A, into the CPS TS. The CRDA is the design basis accident for the subject TS changes. In order to minimize the impact of a CRDA, the BPWS process was developed to minimize control rod reactivity worth for boiling water reactor plants. The proposed improved BPWS further simplifies the shutdown control rod insertion process, and in order to evaluate it, the NRC followed the guidelines of Standard Review Plan Section 15.4.9, and referred to General Design Criterion 28 of Appendix A to 10 CFR Part 50 as its regulatory requirement. The TSTF stated the improved BPWS provides the following benefits: (1) Allows the plant to reach the all-rods-in condition prior to significant reactor cool down, which reduces the potential for recriticality as the reactor cools down; (2) reduces the potential for an operator reactivity control error by reducing the total number of control rod manipulations; (3) minimizes the need for manual scrams during plant shutdowns, resulting in less wear on control rod drive (CRD) system components and CRD mechanisms; and (4) eliminates unnecessary control rod manipulations at low power, resulting in less wear on reactor manual control and CRD system components. The addition of procedural requirements and verifications specified in NEDO-33091-A, along with the proper use of the BPWS will prevent a CRDA from occurring while power is below the low power setpoint (LPSP). The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the 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: Tamra Domeyer, Associate General Counsel, Exelon Generation Company, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: David J. Wrona.

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

Date of amendment request: July 25, 2018. A publicly-available version is in ADAMS under Accession No. ML18206A545.

Description of amendment request: The amendment would revise the TMI-1 Renewed Facility Operating License (RFOL) and associated Technical Specifications (TSs) to the Permanently Defueled Technical Specifications (PDTSs), consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. By letter dated June 20, 2017 (ADAMS Accession No. ML17171A151), Exelon provided formal notification to the NRC of Exelon's contingent determination to permanently cease operations at TMI-1 no later than September 30, 2019. The amendment would eliminate those TSs applicable in operating mode or modes where fuel is placed in the reactor vessel. The amendment would change other TS limiting conditions for operation (LCOs), definitions, surveillance requirements, and administrative controls, as well as several license conditions. The amendment would also modify the licensing basis mitigation strategies for flood mitigation and aircraft impact protection in the air intake tunnel.

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

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

   Response: No.

   The proposed changes would not take effect until TMI has certified to the NRC that it has permanently ceased operation and entered a permanently defueled condition. Because the 10 CFR Part 50 license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR Part 50.82(a)(1) submitted, as specified in 10 CFR Part 0.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible.

   The remaining UFSAR [Updated Final Safety Analysis Report] Chapter 14 postulated design basis accident (DBA) events that could potentially occur at a permanently defueled facility would be a Fuel Handling Accident (FHA) in the Spent Fuel pool (SFP), Waste Gas Tank Rupture (WGTR), and Fuel Cask Drop Accident (FCDA). The FHA analyses for TMI shows that, following 60 days of decay time after reactor shutdown and provided the SFP water level requirements of proposed TS LCO 3/4.1.1 are met, the dose consequences are acceptable without relying on SSCs [structures, systems, and components] to remain functional for accident mitigation during and following the event. The one exception to this is the continued function of the passive SFP structure. The remaining DBAs that support permanently shutdown and defueled condition do not rely on any active safety system for mitigation.

   The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition and safe storage and handling of fuel will be the only operations performed, and therefore, bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

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

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

Response: No.

The proposed changes to delete and/or modify certain [requirements of the] TMI RFOL, TS, or CLB [Current Licensing Basis] have no impact on facility SSCs affecting the safe storage of spent irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of spent irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor, or only to the prevention, diagnosis, or mitigation of reactor related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shutdown and defueled and TMI will no longer be authorized to operate the reactor.

The proposed modification or deletion of requirements of the TMI RFOL, TS, and CLB [does] not affect systems credited in the accident analysis for the remaining credible DBAs at TMI. The proposed RFOL and PDTS will continue to require proper control and monitoring of safety significant parameters and activities. The TS regarding SFP water level and spent fuel storage is retained to preserve the current requirements for safe storage of irradiated fuel.

The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding, spent fuel racks, SFP integrity, and SFP water level). Since extended operation in a defueled condition and safe fuel handling will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

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 proposed changes involve deleting and/or modifying certain [requirements of the] RFOL, TS, and CLB once the TMI facility has been permanently shutdown and defueled. Because the 10 CFR Part 50 license for TMI [will] no longer [authorize] operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR Part 50.82(a)(1) submitted, as specified in 10 CFR Part 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. The remaining postulated DBA events that could potentially occur at a permanently defueled facility would be a FHA, WGTR, and FCDA. The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses.

The proposed changes are limited to those portions of the RFOL, TS, and CLB that are not related to the safe storage of irradiated fuel. The requirements that are proposed to be revised or deleted from the RFOL, TS, and CLB are not credited in the existing accident analysis for the remaining applicable postulated accidents; and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated design basis accidents involving the reactor will no longer be possible because the reactor will be permanently shutdown and defueled and TMI will no longer be authorized to operate the reactor.

Therefore, the proposed changes do 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: Tamra Domeyer, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: James G. Danna.

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

Date of amendment request: September 27, 2018. A publicly-available version is in ADAMS under Accession No. ML18271A009.

Description of amendment request: The amendment would modify the applicability for Technical Specification (TS) Section 3.3.6.2, "Secondary Containment Isolation Instrumentation," Functions 3 and 4, related to reactor building and refueling floor ventilation exhaust, respectively. This change would be implemented in the fall of 2019.

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 requested changes to TS Section 3.3.6.2 to revise the applicability of Functions 3 and 4 as proposed does not eliminate the design function associated with the radiation monitoring instrumentation. The Secondary Containment Isolation Instrumentation will continue to automatically initiate closure of appropriate Secondary Containment Isolation Valves (SCIVs) and start the Standby Gas Treatment (SGT) system as designed to limit fission product release during any postulated Design Basis Accidents (DBAs). These systems are not accident initiators. The proposed changes will continue to assure that these systems perform their design functions, which includes mitigating accidents. The proposed changes do not alter the physical design of any plant Structure, System, or Components (SSC); therefore, the proposed changes have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to DBAs does not change and remains as analyzed in the Updated Final Safety Analysis Report (UFSAR).

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 accident previously evaluated?

   Response: No.

   The requested changes to TS Section 3.3.6.2 to revise the applicability of Functions 3 and 4 as proposed does not adversely affect the design function associated with the radiation monitoring instrumentation. The proposed changes do not change any system operations or maintenance activities that would create the possibility of a new or different kind of accident from one previously evaluated. The Secondary Containment Isolation Instrumentation and SGT system will continue to function as designed. The proposed changes will continue to assure that these systems perform their design functions, which includes mitigating accidents. The proposed changes do not create new failure modes or mechanisms and no new accident precursors are created. The proposed changes do not alter the plant configuration (no new or different type of equipment is being installed) or require any new or unusual Operator actions. The proposed changes do not alter the safety limits or safety analysis assumptions associated with the operation of the plant. The proposed changes do not introduce any new failure modes or mechanisms that could result in a new accident. The proposed changes do not reduce or adversely affect the capabilities of any plant SSC in the performance of their safety function. Also, the response of the plant and the Operators following any DBA is unaffected by the proposed changes.

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

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

   Response: No.

   The requested changes to TS Section 3.3.6.2 to revise the applicability of Functions 3 and 4 as proposed does not alter the design capability associated with the radiation monitoring instrumentation. The proposed changes have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to DBAs does not

> change. The proposed changes do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analyses. There is no change being made to safety analysis assumptions, safety limits or limiting safety system settings that would adversely affect plant safety as a result of the proposed changes.
>
> Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: James G. Danna.

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

Date of amendment request: September 20, 2018. A publicly-available version is in ADAMS under Accession No. ML18263A199.

Description of amendment request: The amendment would make administrative changes to Technical Specification 4.4.2.1, "Inservice Tendon Surveillance Requirements." The amendment would add the words "except where an alternative, exemption, or relief has been authorized by the NRC" to allow NRC-approved exceptions to the 10 CFR 50.55a requirements. Also, the amendment would add a note to exempt from the requirements of Surveillance Requirement 4.0.1.

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 addition of the words "except where an alternative, exemption, or relief has been authorized by the NRC" to Technical Specification (TS) 4.4.2.1 ("Inservice Tendon Surveillance Requirements") and the addition of the wording "The surveillance interval extension allowed per Surveillance Requirement 4.0.1 is not permitted" are administrative changes that have no impact on the accidents analyzed and are not an accident initiator. Since the changes do not impact any conditions that would initiate an accident, the probability or consequences of previously analyzed events is not increased.

   The proposed changes do not involve the modification of any plant equipment or affect plant operation. The proposed changes will have no impact on any safety-related structures, systems, or components.

   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.

   No safety-related equipment, safety function, or plant operation will be altered as a result of these proposed administrative changes. No new operator actions are created as a result of the proposed changes. These administrative changes have no impact on the accidents analyzed in the Updated Final Safety Analysis Report (UFSAR) and are not accident initiators. These proposed changes do not impact the U.S. Nuclear Regulatory Commission Staff's authority to review and grant exceptions. The addition of the wording "The surveillance interval extension allowed per Surveillance Requirement 4.0.1 is not permitted" has been added to address the concerns identified in the U.S. Nuclear Regulatory

Commission's Safety Evaluation Report [(Reference 3 of the licensee's letter dated September 20, 2018)].

Since these proposed changes do not impact any conditions that would initiate an accident, there is no possibility of a new or different kind of accident resulting from these changes. 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 proposed administrative changes do not affect any margins of safety. The margins of safety presently provided by the Technical Specifications remain unchanged. The proposed amendment does not affect the design of the facility or system operating parameters, does not physically alter safety-related systems, structures, or components (SSCs) and does not affect the method in which safety-related systems perform their functions.

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

NRC Branch Chief: James G. Danna.

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1 (FCS), Washington County, Nebraska

Date of amendment request: September 28, 2018. A publicly-available version is in ADAMS under Accession No. ML18275A323.

Description of amendment request: The proposed amendment would revise the Renewed Facility License and the Permanently Defueled Technical Specifications (PDTS) for FCS to reflect the requirements after removal of all remaining spent nuclear fuel from the spent fuel pool (SFP) and its transfer to dry cask storage within an Independent Spent Fuel Storage Installation (ISFSI).

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 amendment would modify the FCS renewed facility operating license and PDTS by deleting the portions of the license and PDTS that are no longer applicable to a facility with no spent nuclear fuel stored in the spent fuel pool, while modifying the remaining portions to correspond to all nuclear fuel stored within an ISFSI. This amendment becomes effective upon removal of all spent nuclear fuel from the FCS SFP and its transfer to dry cask storage within an ISFSI. The definition of safety-related structures, systems, and components (SSCs) in 10 CFR 50.2 states that safety-related SSCs are those relied on to remain functional during and following design basis events to assure:

   1. The integrity of the reactor coolant boundary;
   2. The capability to shutdown the reactor and maintain it in a safe shutdown condition; or
   3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or § 100.11 .

   The first two criteria (integrity of the reactor coolant pressure boundary and safe shutdown of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after all nuclear spent fuel assemblies have been transferred to dry cask storage within an ISFSI, none of the SSCs at FCS are required to be relied on for accident mitigation. Therefore, none of the SSCs at FCS meet the definition of a safety-related SSCs stated in 10 CFR 50.2. The proposed deletion of requirements in the FCS PDTS does not affect systems credited in any accident analysis at FCS.

Chapter 14 of the FCS Defueled Safety Analysis Report (DSAR) described the design basis accident related to the SFP. These postulated accidents are predicated on spent fuel being stored in the SFP. With the removal of the spent fuel from the SFP, there are no remaining spent fuel assemblies to be monitored and there are no credible accidents that require the actions of a Shift Manager, Certified Fuel Handler, or a Non-certified Operator to prevent occurrence or mitigate the consequences of an accident associated with nuclear fuel. The proposed changes do not have an adverse impact on the remaining decommissioning activities or any of their postulated consequences. The proposed changes related to the relocation of certain administrative requirements do not affect operating procedures or administrative controls that have the function of preventing or mitigating any accidents applicable to the safe management of irradiated fuel or decommissioning of the facility. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes eliminate the operational requirements and certain design requirements associated with the storage of the spent fuel in the SFP, and relocate certain administrative controls to the Quality Assurance Topical Report which is a licensee-controlled document. After the removal of the spent fuel from the SFP and transfer to the ISFSI, there are no spent fuel assemblies that remain in the SFP. Coupled with a prohibition against storage of fuel in the SFP, the potential for fuel related accidents is removed. The proposed changes do not introduce any new failure modes.

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

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

Response: No.

The removal of all spent nuclear fuel from the SFP into storage in casks within an ISFSI, coupled with a prohibition against future storage of fuel within the SFP, removes the potential for fuel related accidents.

The design basis and accident assumptions within the FCS DSAR and the PDTS relating to safe management and safety of spent fuel in the SFP are no longer applicable. The proposed changes do not affect remaining plant operations, systems, or components supporting decommissioning activities.

The requirements for SSCs that have been deleted from the FCS PDTS are not credited in the existing accident analysis for any applicable postulated accident; and as such, do not contribute to the margin of safety associated with the accident analysis.

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: Stephen M. Bruckner, Attorney, Fraser Stryker PC LLO, 500 Energy Plaza, 409 South 17th Street, Omaha, NE 68102.

NRC Branch Chief: Bruce A. Watson.

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

Date of amendment request: September 27, 2018. A publicly-available version is in ADAMS under Accession No. ML18270A360.

Description of amendment request: The proposed amendment would correct a non-conservative Technical Specification (TS) 3/4.8.2, “D.C. [Direct Current] Sources – Operating,” by revising the inter-cell resistance value listed in Surveillance Requirements (SRs) 4.8.2.1.b.2 and 4.8.2.1.c.3.

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

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

   Response: No.

   Performing the proposed changes in battery parameter surveillance testing and verification is not a precursor of any accident previously evaluated. Furthermore, these changes will help to ensure that the voltage and capacity of the batteries is such that they will provide the power assumed in calculations of design basis accident mitigation. Therefore, SCE&G concludes that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes to the VCSNS TS SR do not involve any physical modification of the plant or how the plant is operated. No new or different type of equipment will be installed. The proposed changes involve surveillance testing and verification activities. No new failure modes/effects which could lead to an accident whose consequences exceed the consequences of accidents previously analyzed will be introduced by the changes to the TS SR.

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

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

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of the fuel cladding, reactor coolant, and containment systems will not be impacted by the proposed changes.

The proposed VCSNS revisions of the SRs ensure the continued availability and operability of the batteries. As such, sufficient DC capacity to support operation of mitigation equipment remains within the design basis. Therefore, SCE&G concludes that the proposed changes do 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: Kathryn M. Sutton, Morgan, Lewis & Bockius LLP, 1111 Pennsylvania Avenue, NW, Washington, DC 20004.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: October 8, 2018. A publicly-available version is in ADAMS under Accession No. ML18281A014.

Description of amendment request: The proposed amendment would revise the Surveillance Requirement (SR) of Technical Specification (TS) 4.4.6.2.2 (a) to allow the reactor coolant system (RCS) pressure isolation valve (PIV) leakage test to be extended to a performance-based frequency not to exceed 3 refueling outages (RFOs) or 60 months following two consecutive satisfactory tests.

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

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

   Response: No.

   The proposed change involves revising the VCSNS Unit 1, TS wording to reflect a performance-based surveillance testing interval for leakage testing of the RCS PIVs. Specifically, the proposed change revises TS surveillance requirement (SR) 4.4.6.2.2.a to test the RCS PIVs at a frequency from each RFO to a maximum of every third RFO or 60 months by verifying that each of the PIVs tested in the associated RFO based on performance are within the TS allowable leakage limits. The RCS PIVs are defined as two normally closed valves in series with the reactor coolant pressure boundary (RCPB), which separate the high-pressure RCS from an attached lower pressure system. Excessive PIV leakage could lead to overpressure of the low-pressure piping or components, potentially resulting in a LOCA [loss-of-coolant accident] outside of containment.

   TS SR 4.4.6.2.2.a for RCS PIVs provides added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent ISLOCA [intersystem loss-of-coolant accident]. The RCS PIV allowable leakage limit applies to each individual valve. This proposed change does not revise any of the TS RCS PIV allowable leakage limits. In addition, the RCS PIVs will continue to be tested per the VCSNS Inservice Testing Program in accordance with Title 10, *Code of Federal Regulations* (CFR), Section 50.55a, “Codes and standards.” The activity does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. By transitioning to a performance-based leakage testing interval, these valves will continue to be demonstrated operationally ready and reliable. In the event of a PIV leakage test failure, PIV testing would require the component to return to the initial interval of every RFO until good performance is re-established. Therefore, there is no impact on the assurance that the RCS PIVs will be able to perform their safety function(s).

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

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

Response: No.

The proposed change involves revising the VCSNS TS wording to reflect a performance-based surveillance testing interval for leakage testing of the RCS PIVs from each RFO to a maximum of every third RFO or 60 months based on valve performance. The technical testing methodology and associated acceptance criteria remain unchanged. The change in the testing frequency is a performance-based approach, which has been demonstrated acceptable in numerous applications across the industry (RCS PIV testing, 10 CFR 50, Appendix J, Option B).

The testing requirements involved to periodically demonstrate the integrity of the RCS PIVs exist to ensure the plant's ability to mitigate the consequences of an accident. There are not any accident initiators or precursors affected by this change. The proposed TS change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change involves revising the TS SR 4.4.6.2.2.a and associated TS Bases to reflect a performance-based surveillance testing frequency of the RCS PIVs from each RFO to a maximum of every third RFO or 60 months. The technical testing methodology and associated TS allowable leakage limits/acceptance criteria remain unchanged. The testing frequency uses a performance based approach, which has been demonstrated acceptable in numerous applications across the industry (RCS PIV testing, 10 CFR 50, Appendix J, Option B). Thus, this amendment request does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The RCS PIVs will continue to be tested per the VCSNS Inservice Testing Program in accordance with 10 CFR 50.55a.

The primary reason for performance-based PIV test intervals is to eliminate unnecessary thermal cycles. The VCSNS program for monitoring fatigue due to operational cycles and transients consists of review, evaluation, and documentation of RCS operational transients/cycles based on recorded plant operating parameters (i.e., temperature, pressure, flow) for compliance with Technical Specification Sections 3.5.2, 3.5.3, and 5.7.1.

An additional reason for requesting performance-based PIV test intervals is dose reduction to conform with NRC and industry As Low As Reasonably Achievable (ALARA) radiation dose principles. The nominal fuel cycle lengths at VCSNS, Unit 1, are 18 months. However, since RFOs may be scheduled slightly beyond 18 months, a 60-month period is used to provide a bounding timeframe to encompass three RFOs. The review of recent historical data identified that PIV testing each RFO results in a total personnel dose of approximately 300 millirem (milli-Roentgen Equivalent Man, or mrem). Assuming all of the PIVs remain classified as good performers, the proposed extended test intervals would provide for a savings of approximately 600 mrem over an approximate 60-month period (three RFOs).

The proposed surveillance interval extension for the RCS PIVs is based on the performance of the PIVs. The proposed TS change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The design, operation, testing methods, and acceptance criteria for the RCS PIV testing specified in applicable codes and standards will continue to be met.

Therefore, the proposed TS 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: Kathryn M. Sutton, Morgan, Lewis & Bockius LLP, 1111 Pennsylvania Avenue, NW, Washington, DC 20004.

NRC Branch Chief: Michael T. Markley.

Southern Nuclear Operating Company, Inc., Docket No. 52-025, Vogtle Electric Generating Plant (VEGP), Unit 3, Burke County, Georgia

Date of amendment request: October 19, 2018. A publicly-available version is in ADAMS under Accession No. ML18292A660.

Description of amendment request: The requested amendment proposes to depart from certified AP1000 Design Control Document (DCD) Tier 2* material that has been incorporated into the Updated Final Safety Analysis Report (UFSAR). Specifically, the proposed departure consists of changes to Tier 2* information in the UFSAR (which includes the plant-specific DCD information) to change the vertical reinforcement information provided in the VEGP Unit 3 column line 1 wall from elevation 135'-3" to 137'-0".

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.

   As described in UFSAR Subsection 3H.5.1.1, the exterior wall at column line 1 (Wall 1) is located at the south end of the auxiliary building. It is a reinforced concrete wall extending from the basemat at elevation 66'-6" to the roof at elevation 180'-0".

Deviations were identified in the constructed wall from the design requirements. The proposed change modifies the vertical reinforcement information provided in the VEGP Unit 3 Wall 1 from elevation 135'-3" to 137'- 0". This change maintains conformance to the [American Concrete Institute (ACI)] 318-11 and ACI 349-01 codes and has no adverse impact on the seismic response of Wall 1. Wall 1 continues to withstand the design basis loads without loss of structural integrity or the safety-related functions. The proposed change does not affect the operation of any system or equipment that initiates an analyzed accident or alter any SSC [structures, systems, and components] accident initiator or initiating sequence of events.

This change does not adversely affect the design function of the VEGP Unit 3 Wall 1 or the SSCs contained within the auxiliary building. This change does not involve any accident initiating components or events, thus leaving the probabilities of an accident unaltered.

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 modifies the vertical reinforcement information provided in the VEGP Unit 3 Wall 1 from elevation 135'-3" to 137'-0". As demonstrated by the continued conformance to the applicable codes and standards governing the design of the structures, the wall withstands the same effects as previously evaluated. The proposed change does not affect the operation of any systems or equipment that may initiate a new or different kind of accident, or alter any SSC such that a new accident initiator or initiating sequence of events is created. The proposed change does not adversely affect the design function of the auxiliary building Wall 1 or any other SSC design functions or methods of operation in a manner that results in a new failure mode, malfunction, or sequence of events that affect safety-related or non-safety-related equipment. This change does not allow for 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.

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 change modifies the vertical reinforcement information provided in the VEGP Unit 3 Wall 1 from elevation 135'-3" to 137'-0". This change maintains conformance to the ACI 318-11 and ACI 349-01 codes. The change to the vertical reinforcement elevation 135'-3" to 137'-0" does not change the performance of the affected portion of the auxiliary building for postulated loads. The criteria and requirements of ACI 349-01 provide a margin of safety to structural failure. The design of the auxiliary building structure conforms to criteria and requirements in ACI 349-01 and therefore, maintains the margin of safety. The change does not alter any design function, design analysis, or safety analysis input or result, and sufficient margin exists to justify departure from the Tier 2* requirements for the wall. As such, because the system continues to respond to design basis accidents in the same manner as before without any changes to the expected response of the structure, no safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes. Accordingly, no significant safety margin is reduced by the change.

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

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

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

NRC Branch Chief: Jennifer L. Dixon-Herrity.

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

Date of amendment request: October 11, 2018. A publicly-available version is in ADAMS under Accession Nos. ML18284A447.

Description of amendment request: The requested amendment proposes changes to plant-specific Design Control Document (DCD) Tier 2 information in the Updated Final Safety Analysis Report (UFSAR) that involve changes to combined license (COL) Appendix C, and corresponding changes to plant-specific Tier 1 information. The changes would revise the COL to relocate the power operated relief valves in the COL Appendix C, Inspections, Tests, Analyses, and Acceptance Criteria and in the UFSAR. An initial *Federal Register* notice was published on September 19, 2018 (83 FR 47375), providing an opportunity to comment, request a hearing, and petition for leave to intervene for a License Amendment Request (LAR) for the VEGP COLs. The licensee has submitted a revision, dated October 11, 2018, to the original LAR that was dated August 10, 2018. This revision increases the scope of the original LAR. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is also requested for the plant-specific DCD Tier 1 departures.

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

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

   Response: No.

The proposed changes do not affect the operation or reliability of any system, structure or component (SSC) required to maintain a normal power operating condition or to mitigate anticipated transients without safety-related systems. With the proposed changes, the PORV [Power Operated Relief Valve] block valves are still able to perform the safety-related functions of containment isolation, steam generator isolation, and steam generator relief isolation. There is no change to the PORV block valves safety class or safety-related functions.

The relocation of the branch line in which the PORV block valves are installed in allows the PORV block valves to be closer to the containment penetration and maintain compliance with General Design Criterion (GDC) 57 for locating containment isolation valves as close to the containment as practical.

There is no impact to Chapter 15 evaluations. Changes to the PORV block valve and line size do not impact the mass releases to the atmosphere during a Steam Generator Tube Rupture accident. The mass release is limited by the PORV which is more restrictive than the PORV block valve and line size.

There is no impact to any assumed leakage through the PORV line. The existing 12-inch PORV has a design function to limit leakage through the PORV line. Increasing the PORV block valve to 12 inches will increase the leakage through the PORV block valve however it will be that same leakage rate as the 12-inch PORV. Therefore, the leakage rate through the PORV line does not increase and there is no impact to radiation doses.

There is no impact to the assumptions or analysis in the completed safety analysis for radiation doses as a result of the change.

There is no impact to the conclusions of the Pipe Rupture Hazard Analysis (PRHA) because the PORV line is Break Exclusion Zone (BEZ) piping. The proposed changes do not result in any new postulated break locations. Updated analyses confirm that the integrity of the wall adjacent to the MCR [main control room] is unaffected by a postulated main steam line break that causes the PORV line to impact the wall.

There is no change to the valve motor operator. The current motor operator is sufficient to operate the new 12-inch globe valve. Therefore, there is no impact to the Class 1E dc [direct current] and UPS [uninterruptable power supply] System (IDS) battery sizing. There is no change to the valve stroke time, therefore there is no impact to valve open/closure times.

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

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

   Response: No.

   The proposed changes do not affect the operation of systems or equipment that could initiate a new or different kind of accident, or alter any SSC such that a new accident initiator or initiating sequence of events is created. With the proposed changes, the PORV block valves are still able to perform the safety related functions of containment isolation, steam generator isolation, and steam generator relief isolation. There is no change to the PORV block valves safety class or safety-related functions.

   The relocation of the branch line in which the PORV block valves are installed in allows the PORV block valves to be closer to the containment penetration and maintain compliance with General Design Criterion (GDC) 57 for locating containment isolation valves as close to the containment as practical.

   There is no impact to Chapter 15 evaluations. Changes to the PORV block valve and line size do not impact the mass releases to the atmosphere during a Steam Generator Tube Rupture accident. The mass release is limited by the PORV which is more restrictive than the PORV block valve and line size.

   There is no impact to any assumed leakage through the PORV line. The existing 12-inch PORV has a design function to limit leakage through the PORV line. Increasing the PORV block valve to 12 inches will increase the leakage through the PORV block valve however it will be that same leakage rate as the 12-inch PORV. Therefore, the leakage rate through the PORV line does not increase and there is no impact to radiation doses.

   There is no impact to the assumptions or analysis in the completed safety analysis for radiation doses as a result of the change.

   There is no impact to the conclusions of the Pipe Rupture Hazard Analysis (PRHA) because the PORV line is Break Exclusion Zone (BEZ) piping. The proposed changes do not result in any new postulated break locations. Updated analyses confirm that the integrity of the wall adjacent to the MCR is unaffected by a postulated main steam line break that causes the PORV line to impact the wall.

There is no change to the valve motor operator. The current motor operator is sufficient to operate the new 12-inch globe valve. Therefore, there is no impact to the Class 1E dc and UPS System (IDS) battery sizing. There is no change to the valve stroke time, therefore there is no impact to valve open/closure times.

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

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

Response: No.

The proposed changes do not affect existing safety margins. With the proposed changes, the PORV block valves are still able to perform the safety-related functions of containment isolation, steam generator isolation, and steam generator relief isolation. There is no change to the PORV block valves safety class or safety-related functions.

The relocation of the branch line in which the PORV block valves are installed in allows the PORV block valves to be closer to the containment penetration and maintain compliance with General Design Criterion (GDC) 57 for locating containment isolation valves as close to the containment as practical.

There is no impact to Chapter 15 evaluations. Changes to the PORV block valve and line size do not impact the mass releases to the atmosphere during a Steam Generator Tube Rupture accident. The mass release is limited by the PORV which is more restrictive than the PORV block valve and line size.

There is no impact to any assumed leakage through the PORV line. The existing 12-inch PORV has a design function to limit leakage through the PORV line. Increasing the PORV block valve to 12 inches will increase the leakage through the PORV block valve however it will be that same leakage rate as the 12-inch PORV. Therefore, the leakage rate through the PORV line does not increase and there is no impact to radiation doses.

There is no impact to the assumptions or analysis in the completed safety analysis for radiation doses as a result of the change.

The piping analysis for the affected piping has been revised in accordance with the requirements of the UFSAR. All stresses and interface loads remain acceptable and within the limits described in the UFSAR. The piping support calculations have been revised using the load combinations prescribed in the UFSAR, and the critical interaction ratio for each support is less than 1.0; therefore, a positive design margin exists. The proposed changes did not affect any of the piping packages chosen (as listed in the UFSAR) to demonstrate piping design for piping design acceptance criteria closure. There is no impact to the conclusions of the Pipe Rupture Hazard Analysis (PRHA) because the PORV line is Break Exclusion Zone (BEZ) piping. The proposed changes do not result in any new postulated break locations. Updated analyses confirm that the integrity of the wall adjacent to the MCR is unaffected by a postulated main steam line break that causes the PORV line to impact the wall. The piping and components downstream of the PORV are nonsafety-related and are not affected by this activity.

The structural concrete floors and walls which make up the bounds of the affected rooms were analyzed for the downstream impacts due to the proposed changes. The results conclude that the applicable acceptance criteria of the UFSAR are met. All applicable load combinations shown in the UFSAR were considered. Critical sections defined in the UFSAR within the scope of analysis remain unchanged along with the typical reinforcement configuration presented in the UFSAR. Therefore, all structural evaluations are within the bounds of the acceptance criteria and meet the licensing requirements imposed in the UFSAR.

There is no change to the valve motor operator. The current motor operator is sufficient to operate the new 12-inch globe valve. Therefore, there is no impact to the Class 1E dc and UPS System (IDS) battery sizing. There is no change to the valve stroke time, therefore there is no impact to valve open/closure times.

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

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

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

NRC Branch Chief: Jennifer L. Dixon-Herrity.

Tennessee Valley Authority (TVA), Docket No. 50-391, Watts Bar Nuclear Plant (WBN), Unit 2, Rhea County, Tennessee

Date of amendment request: May 14, 2018. A publicly available version is in ADAMS under Accession No. ML18138A232.

Description of amendment request: The proposed amendment would modify the WBN, Unit 2, Technical Specification (TS) 5.7.2.12, "Steam Generator (SG) Program," and TS 5.9.9, "Steam Generator Tube Inspection Report," to use the voltage-based alternate repair criteria (ARC) specified in the guidelines contained in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

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.

   Allowing the use of alternate repair criteria as proposed in this amendment request does not involve a significant increase in the probability or consequence of an accident previously evaluated.

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the TSP [tube support plates]. Test data indicates that tube burst cannot occur within the TSP, even for tubes, which have 100% through-wall electric discharge machining (EDM) notches, 0.75 inches long, provided that the TSP is adjacent to the notched area. Because tube-to-tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics, which maintain a margin of safety of 1.4 times the bounding faulted condition [i.e., main steam line break (MSLB)] differential pressure of 2405 psig. GL 95-05 recommends that maintenance of a safety factor of 1.4 times the MSLB pressure differential, consistent with the structural limits in Regulatory Guide (RG) 1.121, on tube burst is satisfied by 3/4-inch diameter tubing with bobbin coil indications with signal amplitudes less than the tube structural limit (V_{SL}) of 6.03 volts, regardless of the indicated depth measurement. At the FDB [flow distribution baffles], a safety factor of three against the normal operating condition ΔP is applied. A voltage of V_{SL} = 3.81 volts satisfies the burst capability recommendation at the FDB.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently approved NRC database to determine the V_{SL}. The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_G) to establish V_{URL}.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valves (MSIVs) represents the most limiting radiological condition relative to the alternate voltage-based repair criteria. In support of implementation of the revised repair limit, TVA will determine whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary to secondary leakage would result in site boundary doses within a fraction of the 10 CFR 100 guidelines or control room doses within the 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 limit. A separate calculation has determined this allowable MSLB leakage limit to be four gallons per minute (gpm) in the faulted loop.

The methods for calculating the radiological dose consequences for this postulated MSLB are consistent with the WBN dual-unit Updated Final Safety Analysis Report (UFSAR) Chapter 15.

In summary, the calculated radiological consequences in the control room and at the exclusion area boundary and the low population zone are in compliance with the guidelines in the Standard Review Plan, Chapter 15, and the regulations in 10 CFR 50, Appendix A, GDC 19, and 10 CFR 100 reported for the postulated steamline break event. Therefore, it is concluded that the proposed changes do not result in a significant increase in the radiological consequences of an accident previously analyzed.

Consistent with the guidance of GL 95-05, Section 2.c, the WBN Unit 2 MSLB leak rate analysis would be performed, prior to returning the SGs to service, based on either the projected next end-of-cycle (EOC) voltage distribution or the actual measured bobbin voltage distribution. The method to be used for the first outage when ODSCC [outside diameter stress corrosion cracking] indication growth rates are available will be based on the indications found during that outage. As noted in GL 95-05, it may not always be practical to complete EOC calculations prior to returning the SGs to service. Under these circumstances, it is acceptable to use the actual measured bobbin voltage distribution instead of the projected EOC voltage distribution to determine whether the reporting criteria are being satisfied.

Therefore, the voltage-based ARC at WBN Unit 2 does not adversely affect SG tube integrity and implementation is shown to result in acceptable radiological dose consequences. Therefore, the proposed TS change does not result in a significant increase in the probability or consequences of an accident previously evaluated within the WBN Unit 2 UFSAR.

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

   Response: No.

   Implementation of the proposed SG tube voltage-based ARC does not introduce any changes to the plant design basis. Neither a single nor multiple tube rupture event would be expected in an SG in which the repair limit has been applied (during all plant conditions).

   The bobbin probe voltage-based tube repair criteria of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100 percent eddy current inspection sample size at the tube support plate elevations, and rotating probe coil (RPC) or equivalent inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

As SG tube integrity upon implementation of the 1.0 volt repair limit continues to be maintained through in-service inspection and primary to secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

Response: No.

The use of the voltage-based bobbin probe tube support plate elevation repair criteria at WBN Unit 2 maintains SG tube integrity commensurate with the guidance of RG 1.121. RG 1.121 describes a method acceptable to the NRC for meeting GDCs 14, 15, and 32 by reducing the probability or the consequences of SG tube rupture. This reduction is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the proposed criteria, even under the worst-case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to an SG tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations is confirmed to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

Implementation of the TSP intersection voltage-based repair criteria will decrease the number of tubes that must be plugged. The installation of SG tube plugs reduces the reactor coolant system flow margin. Thus, implementation of the 1.0 volt repair limit will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

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

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

NRC Branch Chief: Undine S. Shoop.

Tennessee Valley Authority, Docket Nos. 50-390 and 50-391, Watts Bar Nuclear Plant (WBN), Units 1 and 2, Rhea County, Tennessee

Date of amendment request: February 28, 2018. A publicly-available version is in ADAMS under Accession No. ML18060A337.

Description of amendment request: The proposed amendments would modify the WBN, Units 1 and 2, Technical Specification (TS) 3.8.9, to add a new Condition C with an 8-hour completion for performing maintenance on the opposite unit's vital bus when the opposite unit is in Mode 5, Mode 6, or defueled. The proposed change would allow greater operational flexibility for two-unit operation at WBN.

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

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

   Response: No.

   The proposed change modifies the Required Actions for the opposite unit's 120-volt (V) alternating current (AC) vital bus system. This change will not affect the probability of an accident, because the distribution system is not an initiator of any accident sequence analyzed in the UFSAR [updated final safety analysis report]. Rather, the opposite unit's distribution system support equipment is used to mitigate accidents. The consequences of an analyzed accident will not be significantly increased because the minimum requirements for distribution systems will be maintained to ensure the availability of the required power to mitigate accidents assumed in the UFSAR. Operation in accordance with the proposed TS will ensure that sufficient onsite electrical distribution systems are operable as required to support the unit's required features. Therefore, the mitigating functions supported by the onsite electrical distribution systems will continue to provide the protection assumed by the accident analysis. The integrity of fission product barriers, plant configuration, and operating procedures as described in the UFSAR will not be affected by the proposed changes. Thus, the consequences of previously analyzed accidents will not increase by implementing these changes.

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

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

   Response: No.

   The proposed change modifies the Required Actions for the opposite unit’s 120V AC vital bus system. This change will not physically alter the plant (no new or different type of equipment will be installed). The proposed change will maintain the minimum requirements for onsite electrical distribution systems to ensure the availability of the equipment required to mitigate accidents assumed in the UFSAR.

   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 amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change modifies the Required Actions for the opposite unit's 120V AC vital bus system. The margin of safety is not affected by this change because the minimum requirements for onsite electrical distribution systems will be maintained to ensure the availability of the required power to shutdown the reactor and maintain it in a safe shutdown condition after an AOO [anticipated operational occurrence] or a postulated DBA [design-basis accident].

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

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

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

NRC Branch Chief: Undine s. Shoop.

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

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items can be accessed as described in the "Obtaining Information and Submitting Comments" section of this document.

Duke Energy Progress, LLC, Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: June 28, 2017, as supplemented by letters dated July 20 and September 14, 2017; and January 18, February 16, and April 13, 2018.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) for fuel storage criticality to account for the use of neutron absorbing spent fuel pool rack inserts and soluble boron for the purpose of criticality control in the boiling-water reactor storage racks that currently credit Boraflex.

Date of issuance: October 22, 2018.

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

Amendment No.: 167. A publicly-available version is in ADAMS under Accession No. ML18204A286; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF-63: The amendment revised the Renewed Facility Operating License and TSs.

Date of initial notice in *Federal Register*: December 5, 2017 (82 FR 57481). The supplemental letters dated July 20 and September 14, 2017; and January 18, February 16, and April 13, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 2018.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50-397, Columbia Generating Station, Benton County, Washington

Date of amendment request: October 23, 2017, as supplemented by letters dated November 15, 2017, and June 27, 2018.

Brief description of amendment: The amendment replaced the existing Technical Specification (TS) requirements related to "operations with a potential for draining the reactor vessel" (OPDRVs) with new requirements on reactor pressure vessel (RPV) water inventory control to protect Safety Limit 2.1.1.3. Safety Limit 2.1.1.3 requires RPV water level to be greater than the top of active irradiated fuel. The changes are based on NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-542, Revision 2, "Reactor Pressure Vessel Water Inventory Control."

Date of issuance: October 30, 2018.

Effective date: As of its date of issuance and shall be implemented at the beginning of the next refueling outage scheduled for May 2019.

Amendment No.: 251. A publicly-available version is in ADAMS under Accession No. ML18255A350; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF-21: The amendment revised the Renewed Facility Operating License and TS.

Date of initial notice in *Federal Register*: January 16, 2018 (83 FR 2227). The supplemental letter dated June 27, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2018.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit 1 (ANO-1), Pope County, Arkansas

Date of amendment request: October 2, 2017, as supplemented by letters dated April 26 and August 10, 2018.

Brief description of amendment: The amendment revised the ANO-1 Technical Specification (TS) Bases for TS 3.7.5, "Emergency Feedwater (EFW) System," to identify the conditions in which TS 3.7.5, Condition A, 7-day Completion Time (CT) and

Condition C, 24-hour CT should apply to the ANO-1 turbine-driven EFW pump steam supply valves.

Date of issuance: October 24, 2018.

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

Amendment No.: 261. A publicly-available version is in ADAMS under Accession No. ML18260A339; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-51: The amendment revised the TS Bases.

Date of initial notice in *Federal Register*: December 5, 2017 (82 FR 57473). The supplemental letters dated April 26 and August 10, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 2018.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50-219, Oyster Creek Nuclear Generating Station (Oyster Creek), Ocean County, New Jersey

Date of amendment request: November 16, 2017, as supplemented by letter dated March 29, 2018.

Brief description of amendment: The amendment revised the Oyster Creek Renewed Facility Operating License and the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications consistent with the permanent cessation of operations and permanent removal of fuel from the reactor vessel.

Date of issuance: October 26, 2018.

Effective date: The license amendment is effective on November 16, 2018, and shall be implemented in 60 days from the effective date.

Amendment No.: 295. A publicly-available version is in ADAMS under Accession No. ML18227A338; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-16: The amendment revised the Renewed Facility Operating License and TS.

Date of initial notice in *Federal Register*: January 16, 2018 (83 FR 2229). The supplemental letter dated March 29, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 2018.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: June 25, 2018, as supplemented by letter dated August 29, 2018.

Brief description of amendment: The amendment revised the R. E. Ginna Nuclear Power Plant's Technical Specification (TS) 3.1.4, "Rod Group Alignment Limits"; TS 3.1.5, "Shutdown Bank Insertion Limit"; TS 3.1.6, "Control Bank Insertion Limits"; and TS 3.1.7, "Rod Position Indication," consistent with NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-547, Revision 1, "Clarification of Rod Position Requirements," dated March 4, 2016.

Date of issuance: October 31, 2018.

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

Amendment No.: 131. A publicly-available version is in ADAMS under Accession No. ML18295A630; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-18: The amendment revised the Renewed Facility Operating License and TSs.

Date of initial notice in *Federal Register*: July 31, 2018 (83 FR 36976). The supplemental letter dated August 29, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2018.

No significant hazards consideration comments received: No.

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

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

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

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

Date of amendment request: April 25, 2018.

Brief description of amendments: The amendments revised the Technical Specification (TS) requirements for inoperable snubbers for each facility. The amendments also made other administrative changes to the TS.

Date of issuance: October 29, 2018.

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

Amendment Nos.: Clinton – 220 (Unit 1); Dresden – 259 (Unit 2), 252 (Unit 3); LaSalle – 231 (Unit 1), 217 (Unit 2); and Quad Cities – 271 (Unit 1), 266 (Unit 2). A publicly-available version is in ADAMS under Accession No. ML18254A367. Documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. NPF-62, DPR-19, DPR-25, NPF-11, NPF-18, DPR-29, and DPR-30: The amendments revised the Facility Operating Licenses and TS.

Date of initial notice in *Federal Register*: June 19, 2018 (83 FR 28460).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 29, 2018.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units 2 and 3, York County, Pennsylvania

Date of amendment request: August 30, 2017, as supplemented by letters dated October 24, 2017; and May 7, June 6, August 10, and August 22, 2018.

Brief description of amendments: The amendments added a new license condition to the Renewed Facility Operating Licenses to allow the implementation of risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors in accordance with 10 CFR 50.69.

Date of issuance: October 25, 2018.

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

Amendment Nos.: 321 (Unit 2) and 324 (Unit 3). A publicly-available version is in ADAMS under Accession No. ML18263A232; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Renewed Facility Operating Licenses.

Date of initial notice in *Federal Register*: November 21, 2017 (82 FR 55404). The supplemental letters dated May 7, June 6, August 10, and August 22, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 2018.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs), Calvert County, Maryland

Date of amendment request: February 25, 2016, as supplemented by letters dated April 3, 2017, and January 11, January 18, June 21, and August 27, 2018.

Brief description of amendments: The amendments revised the Calvert Cliffs Technical Specifications (TS) related to completion times for required actions to provide the option to calculate longer risk-informed completion times. The amendments also added a new program, the "Risk Informed Completion Time Program," to TS Section 5.5, "Programs and Manuals."

Date of issuance: October 30, 2018.

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

Amendment Nos.: 326 (Unit 1) and 304 (Unit 2). A publicly-available version is in ADAMS under Accession No. ML18270A130; documents related to these amendments are listed in the safety evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-53 and DPR-69: The amendments revised the Renewed Facility Operating Licenses and TS.

Date of initial notice in *Federal Register*: September 4, 2018 (83 FR 44920). The supplemental letter dated August 27, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendments is contained in a safety evaluation dated October 30, 2018.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 2, 2018.

Brief description of amendments: The amendments revised the Technical Specifications (TS) by removing Figure 5.1-1, "Site Area Map"; removing Technical Specification references to Figure 5.1-1; and adding a site description.

Date of issuance: November 2, 2018.

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

Amendment Nos.: 246 (Unit No. 1) and 197 (Unit No. 2). A publicly-available version is in ADAMS under Accession No. ML18274A224; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-67 and NPF-16: The amendments revised the Renewed Facility Operating Licenses and TS.

Date of initial notice in *Federal Register*: August 28, 2018 (83 FR 43905).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 2018.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 10, 2017.

Brief description of amendment: The amendment revised the Technical Specifications (TS) for DAEC to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-551, Revision 3, "Revise Secondary Containment Surveillance Requirements," dated November 10, 2017 (ADAMS Accession No. ML17318A240).

Date of issuance: October 31, 2018.

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

Amendment No.: 307. A publicly-available version is in ADAMS under Accession No. ML18241A383; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-49: The amendment revised the Renewed Facility Operating License and TS.

Date of initial notice in *Federal Register*: February 27, 2018 (83 FR 8517).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2018.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 20, 2017, as supplemented by letters dated June 1 and September 11, 2018.

Brief description of amendment: The amendment revised the Monticello Technical Specification (TS) to adopt Technical Specification Task Force (TSTF) Traveler TSTF-542, "Reactor Pressure Vessel Water Inventory Control."

Date of issuance: October 29, 2018.

Effective date: As of the date of issuance and shall be implemented prior to the next refueling outage.

Amendment No.: 198. A publicly-available version is in ADAMS under Accession No. ML18250A075; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-22. The amendment revised the Renewed Facility Operating License and TS.

Date of initial notice in *Federal Register*: December 19, 2017 (82 FR 60228). The supplemental letters dated June 1 and September 11, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 2018.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50-354, Hope Creek Generating Station (Hope Creek), Salem County, New Jersey

Date of amendment request: September 21, 2017, as supplemented by letters dated June 27, July 19, and September 6, 2018.

Brief description of amendment: The amendment revised the Hope Creek Technical Specifications (TS) by replacing the existing specifications related to “operation with a potential for draining the reactor vessel” with revised requirements for reactor pressure vessel water inventory control to protect Safety Limit 2.1.4. Safety Limit 2.1.4 requires reactor vessel water level to be greater than the top of active irradiated fuel. The amendment adopted changes with variations, as noted in the license amendment request, and is based on the NRC-approved safety evaluation for Technical Specifications Task Force (TSTF) Traveler TSTF-542, Revision 2, “Reactor Pressure Vessel Water Inventory Control,” dated December 20, 2016.

Date of issuance: October 30, 2018.

Effective date: As of the date of issuance and shall be implemented prior to entering Operating Condition 4 for the next Hope Creek refueling outage schedule for fall 2019 (H1R22).

Amendment No.: 213. A publicly-available version is in ADAMS under Accession No. ML18260A203; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License No. NPF-57: The amendment revised the Renewed Facility Operating License and TS.

Date of initial notice in *Federal Register*: January 30, 2018 (83 FR 4294). The supplemental letters dated June 27, July 19, and September 6, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2018.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 12, 2017, as supplemented by letter dated April 5, 2018.

Brief description of amendments: The amendments revised Technical Specification (TS) 5.5.17, "Containment Leakage Rate Testing Program," for Vogtle to (1) increase the existing Type A integrated leakage rate test interval from 10 to 15 years; (2) extend the Type C containment isolation valve leaking testing to a 75-month frequency; (3) adopt the use of American National Standards Institute/American Nuclear Society 56.8-2002, "Containment System Leakage Testing Requirements"; and (4) adopt a more conservative grace interval for Type A, B, and C tests.

Date of issuance: October 29, 2018.

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

Amendment Nos.: 197 (Unit 1) and 180 (Unit 2). A publicly-available version is in ADAMS under Accession No. ML18263A039; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. NPF-68 and NPF-81: The amendments revised the Renewed Facility Operating Licenses and TS.

Date of initial notice in *Federal Register*: December 5, 2017 (82 FR 57474). The supplemental letter dated April 5, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 2018.

No significant hazards consideration comments received: No.

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: April 13, 2018, as supplemented by letter dated August 10, 2018.

Description of amendment: The amendment authorized changes to the VEGP Units 3 and 4 Combined Operating License (COL) Appendix A, Technical Specifications (TS). The amendment authorized departures from associated Updated Final Safety Analysis Report information (which includes the plant specific design control document Tier 2 information) with changes which conform with the authorized TS changes.

Date of issuance: October 11, 2018.

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

Amendment Nos.: 146 (Unit 3) and 145 (Unit 4). A publicly-available version is in ADAMS under Accession No. ML18248A137; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Facility Combined Licenses Nos. NPF-91 and NPF-92: The amendment revised the Facility Combined Licenses and TS.

Date of initial notice in *Federal Register*: June 27, 2018 (83 FR 30199). The supplemental letter dated August 10, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in the Safety Evaluation dated October 11, 2018.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50-391, Watts Bar Nuclear Plant, Unit 2, Rhea County, Tennessee

Date of amendment request: October 11, 2017.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.3.1, Table 3.3.1-1, "Reactor Trip System (RPS) Instrumentation," to increase the values for the nominal trip setpoint and the allowable value for Function 14.a, "Turbine Trip − Low Fluid Oil Pressure." The changes are due to the planned replacement and relocation of the pressure switches from the low pressure auto-stop trip fluid oil header to the high pressure turbine electrohydraulic control (EHC) oil header. The changes are needed due to the higher EHC system operating pressure.

Date of issuance: October 30, 2018.

Effective date: As of the date of issuance and shall be implemented no later than startup from the Unit 2 refueling outage scheduled for spring 2019.

Amendment No.: 22. A publicly-available version is in ADAMS under Accession No. ML18255A156; documents related to the amendment are listed in the Safety Evaluation enclosed with the amendment.

Facility Operating License No. NPF-96: The amendment revised the Facility Operating License and TS.

Date of initial notice in *Federal Register*: March 13, 2018 (83 FR 10924).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2018.

No significant hazards consideration comments received: No.

IV. Notice of Issuance of Amendments to Facility Operating Licenses and Combined Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual notice of consideration of issuance of amendment, proposed no significant hazards consideration determination, and opportunity for a hearing.

For exigent circumstances, the Commission has either issued a *Federal Register* notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 51.12(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 application for amendment, (2) the amendment to Facility Operating License or Combined License, as applicable, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items can be accessed as described in the "Obtaining Information and Submitting Comments" section of this document.

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, any persons (petitioner) whose interest may be affected by this action may file a request for a hearing and petition for leave to intervene (petition) with respect to the action. Petitions shall be filed in accordance with the Commission's "Agency Rules of Practice and Procedure" in 10 CFR part 2. Interested persons should consult a current copy of

10 CFR 2.309. The NRC's regulations are accessible electronically from the NRC Library on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Alternatively, a copy of the regulations is available at the NRC's Public Document Room, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. If a petition is filed, the Commission or a presiding officer will rule on the petition and, if appropriate, a notice of a hearing will be issued.

As required by 10 CFR 2.309(d) the petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements for standing: (1) the name, address, and telephone number of the petitioner; (2) the nature of the petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the 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 petitioner's interest.

In accordance with 10 CFR 2.309(f), the petition must also set forth the specific contentions which the petitioner seeks to have litigated in the proceeding. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner must 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 petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue. The petition must include sufficient information to show that a genuine dispute exists with the applicant or licensee on a material issue of law or fact. Contentions must be limited to matters within the scope of the proceeding. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to satisfy the requirements at 10 CFR 2.309(f) 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. Parties have the opportunity to participate fully in the conduct of the hearing with respect to resolution of that party's admitted contentions, including the opportunity to present evidence, consistent with the NRC's regulations, policies, and procedures.

Petitions must be filed no later than 60 days from the date of publication of this notice. Petitions and motions for leave to file new or amended contentions that are filed after the deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i) through (iii). The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to establish 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 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 the amendment unless the Commission finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR part 2.

A State, local governmental body, Federally-recognized Indian Tribe, or agency thereof, may submit a petition to the Commission to participate as a party under 10 CFR 2.309(h)(1). The petition should state the nature and extent of the petitioner's interest in the proceeding. The petition should be submitted to the Commission no later than 60 days from the date of publication of this notice. The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document, and should meet the requirements for petitions set forth in this section, except that under 10 CFR 2.309(h)(2) a State, local governmental body, or Federally-recognized Indian Tribe, or agency thereof does not need to address the standing requirements in 10 CFR 2.309(d) if the facility is located within its boundaries. Alternatively, a State, local governmental body, Federally-recognized Indian Tribe, or agency thereof may participate as a non-party under 10 CFR 2.315(c).

If a hearing is granted, any person who is not a party to the proceeding and is not affiliated with or represented by a party may, at the discretion of the presiding officer, be permitted to make a limited appearance pursuant to the provisions of 10 CFR 2.315(a). A person making a limited appearance may make an oral or written statement of his or her position on the issues but may not otherwise participate in the proceeding. A limited appearance may be made at any session of the hearing or at any prehearing conference, subject to the limits and conditions as may be imposed by the presiding officer. Details regarding the opportunity to make a limited appearance will be provided by the presiding officer if such sessions are scheduled.

B. Electronic Submissions (E-Filing).

All documents filed in NRC adjudicatory proceedings, including a request for hearing and petition for leave to intervene (petition), 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 that request to participate under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007, as amended at 77 FR 46562; August 3, 2012). 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. Detailed guidance on making electronic submissions may be found in the Guidance for Electronic Submissions to the NRC and on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html*. 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 (1) request a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign submissions and access the E-Filing system for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a petition or other adjudicatory document (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

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

A person filing electronically using the NRC's adjudicatory E-Filing system may seek assistance by contacting the NRC's Electronic Filing Help Desk through the "Contact Us" link located on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail to *MSHD.Resource@nrc.gov*, or by a toll-free call at 1-866-672-7640. The NRC Electronic Filing Help Desk is available between 9 a.m. and 6 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 stating why there is good cause for not filing electronically and 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, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff. Participants filing adjudicatory documents 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 *https://adams.nrc.gov/ehd*, unless excluded pursuant to an order of the Commission or the presiding officer. If you do not have an NRC-issued digital ID certificate as described above, click cancel when the link requests certificates and you will be automatically directed to the NRC's electronic hearing dockets where you will be able to access any publicly available documents in a particular hearing docket. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or personal phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. For example, in some instances, individuals provide home addresses in order to demonstrate proximity to a facility or site. 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.

Vistra Operations Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Nuclear Power Plant (CPNPP), Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: September 5, 2018, as supplemented by letters dated September 20 and October 3, 2018.

Description of amendment: The amendments revised the CPNPP Technical Specification (TS) 3.8.4, "DC [Direct Current] Sources - Operating," by adding a new REQUIRED ACTION to CONDITION B and an extended COMPLETION TIME on a one-time basis to repair two affected battery cells on the CPNPP Unit 1, Train B safety-related batteries.

Date of issuance: October 25, 2018.

Effective date: As of the date of issuance and shall be implemented immediately as of its date of issuance.

Amendment Nos.: Unit 1 - 170; Unit 2 - 170. A publicly-available version is in ADAMS under Accession No. ML18267A384; documents related to the amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. NPF-87 and NPF-89: The amendments revised the Facility Operating Licenses and TS.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes.

The license amendment request was originally noticed in the *Federal Register* on September 18, 2018 (83 FR 47203). Subsequently, by letters dated September 20 and October 3, 2018, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, on October 10, 2018 (83 FR 50971), the NRC published a second proposed NSHC determination, which superseded the original notice in its entirety. This included an individual 14-day notice for comments and provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by December 10, 2018, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a Safety Evaluation dated October 25, 2018.

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

NRC Branch Chief: Robert J. Pascarelli.

Dated at Rockville, Maryland, this 8th day of November 2018.

For the Nuclear Regulatory Commission.

Kathryn M. Brock,
Deputy Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.

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