[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2017-0140]

Biweekly Notice

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

SUMMARY: Pursuant to Section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (NRC) is publishing this regular biweekly notice. The Act requires the Commission to publish notice of any amendments issued, or proposed to be issued, and grants the Commission the authority to issue and make immediately effective any amendment to an operating license or combined license, as applicable, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from May 23, 2017, to June 2, 2017. The last biweekly notice was published on June 6, 2017.

DATES: Comments must be filed by July 19, 2017. A request for a hearing must be filed by August 18, 2017.

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

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2017-0140. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individual listed in the FOR FURTHER INFORMATION CONTACT section of this document.
- **Mail comments to:** Cindy Bladey, Office of Administration, Mail Stop: TWFN-8-D36M, 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: Lynn Ronewicz, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone: 301-415-1927, e-mail: Lynn.Ronewicz@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC-2017-0140 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-2017-0140.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by 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-2017-0140 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 by August 18, 2017. 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.

Entergy Nuclear Operations, Inc., Docket Nos. 50-247 and 50-286, Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3), Westchester County, New York

Date of amendment request: December 14, 2016, as supplemented by letter dated April 19, 2017. Publicly available versions are in ADAMS under Package Accession No. ML16355A066 and Accession No. ML17114A467, respectively.

Description of amendment request: The amendments would revise the Appendix C Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.1.2 for IP2 and IP3 and Appendix A TS LCO 3.7.13 for IP2. These LCOs ensure that the fuel to be loaded into the Shielded Transfer Canister (STC) meets the design basis for the STC and has an acceptable rack location in the IP2 spent fuel pit before the STC is loaded with fuel. The proposed changes to these LCOs would increase the population of IP3 fuel eligible for transfer to the IP2 spent fuel pit and maintain full core offload capability for IP3.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below, with NRC staff's edits in square brackets:

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

   Response: No.

   The proposed amendment would modify the IP2 and IP3 Technical Specifications (TS) to incorporate the results of revised criticality, thermal, and shielding and dose analyses and evaluations.

   [For IP2,] the proposed amendment was evaluated for impact on the following previously evaluated events and accidents: STC Criticality Accidents, SFP Criticality Accidents, Boron Dilution Accidents, Fuel Handling Accidents, Loss of Spent Fuel Pool [SFP] Cooling, and Natural Events.

   [IP2] STC Criticality Accidents

   The STC criticality accident considered were: abnormal temperature, dropped, mislocated, and misloaded fuel assemblies, and misalignment between the active fuel region and the neutron absorber.

   The probability of an STC criticality accident will not increase significantly due to the proposed changes because the individual fuel assemblies will be loaded into the STC in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of an STC criticality accident are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for these accidents.

[IP2] SFP Criticality Accidents

The SFP criticality accident of record considered the following accidents 1.) a dropped fuel assembly or an assembly placed alongside a rack, 2.) a misloaded fuel assembly, and 3.) abnormal heat loads. Because the IP2 and IP3 fuel assemblies are identical [with] regards [to] those parameters that are utilized in the design basis criticality analysis (DBA) to qualify fresh fuel these accidents are bounding for IP3 fuel.

The probability of an SFP criticality accident will not increase significantly due to the proposed changes because the individual fuel assemblies will be loaded into the SFP in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of an SFP criticality accident are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for this accident.

[IP2] STC Thermal Accidents

The thermal analyses demonstrate that the postulated accidents (rupture of the HI-TRAC water jacket, 50-gallon transported fuel tank rupture and fire, simultaneous loss of water from the water jacket and HI-TRAC annulus, fuel misload, hypothetical tipover, and crane malfunction) continue to meet their acceptance criteria.

The probability of an STC thermal accident will not increase significantly because the individual fuel assemblies will be loaded into the SFP in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of an STC thermal accident will not increase significantly because the thermal analysis demonstrates that the same thermal acceptance criteria and requirements continue to be met for this accident.

[IP2] Boron Dilution Accident

The probability of a boron dilution event remains the same because the proposed change does not alter the manner in which the IP2 spent fuel cooling system or any other plant system is operated, or otherwise increase the likelihood of adding significant quantities of unborated water into the spent fuel pit.

The consequences of the boron dilution event remains the same. The reactivity of the STC filled with the most reactive combination of approved fuel assemblies in unborated water results in a k_{eff} less than 0.95. Thus, even in the unlikely event of a complete dilution of the spent fuel pit water, the STC will remain safely subcritical.

[IP2] Fuel Handling Accident

The probability of an FHA will not increase significantly due to the proposed changes because the individual fuel assemblies will be moved between the STC and the spent fuel pit racks and the STC and HI-TRAC will be moved in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of the existing fuel handling accident remain bounding because the IP3 fuel assembly design is essentially the same as the IP2 design and the IP3 fuel assemblies to be transferred to IP2 will be cooled a minimum of 6 years. This compares with a cooling time of 84 hours used in the existing FHA radiological analysis. The 6-year cooling time results in a significant reduction in the radioactive source term available for release from a damaged fuel assembly compared to the source term considered in the design basis FHA radiological analysis. The consequences of the previously analyzed fuel assembly drop accident, therefore, continue to provide a bounding estimate of offsite dose for this accident.

[IP2] Loss of Spent Fuel Pool Cooling

The probability of a loss of spent fuel pit cooling remains the same because the proposed change does not alter the manner in which the IP2 spent fuel cooling loop is operated, designed or maintained.

The consequences of a loss of spent fuel pit cooling remains the same because the thermal design basis for the spent fuel pit cooling loop provides for all fuel pit rack locations to be filled at the end of a full core discharge and therefore the design basis heat load effectively includes any heat load associated with the assemblies within the STC.

[IP2] Natural Events

The natural events considered include the following accidents 1.) a seismic event, 2.) high winds, tornado and tornado missiles, 3.) flooding and 4) a lightning strike.

The probability of natural event will not increase due to the proposed changes because there are no elements of the proposed changes that influence the occurrence of any natural event.

The consequences of a natural event will not increase due to the proposed changes because the structural analyses design limits continue to be met. A lightning strike may cause ignition of the VCT fuel but this event is addressed under STC thermal accidents.

[For IP3,] the proposed amendment was evaluated for impact on the following previously evaluated events and accidents: STC Criticality Accidents, SFP Criticality Accidents, Boron Dilution Accidents, Fuel Handling Accidents, Loss of Spent Fuel Pool Cooling, and Natural Events.

[IP3] STC Criticality Accidents

The STC criticality accident considered were: abnormal temperature, dropped, mislocated, and misloaded fuel assemblies, and misalignment between the active fuel region and the neutron absorber.

The probability of an STC criticality accident will not increase significantly due to the proposed changes because the individual fuel assemblies will be loaded into the STC in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of an STC criticality accident are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for these accidents.

[IP3] STC Thermal Accidents

The thermal analyses demonstrate that the postulated accidents (rupture of the HI-TRAC water jacket, 50-gallon transported fuel tank rupture and fire, simultaneous loss of water from the water jacket and HI-TRAC annulus, fuel mislead, hypothetical tipover, and crane malfunction) continue to meet their acceptance criteria. The probability of an STC thermal accident will not increase significantly because the individual fuel assemblies will be loaded into the SFP in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of an STC thermal accident will not increase significantly because the thermal analysis demonstrates that the same thermal acceptance criteria and requirements continue to be met for this accident.

[IP3] Boron Dilution Accident

The probability of a boron dilution event remains the same because the proposed change does not alter the manner in which the IP3 spent fuel cooling system or any other plant system is operated, or otherwise

increase the likelihood of adding significant quantities of unborated water into the spent fuel pit.

The consequences of the boron dilution event remains the same. The reactivity of the STC filled with the most reactive combination of approved fuel assemblies in unborated water results in a k_{eff} less than 0.95. Thus, even in the unlikely event of a complete dilution of the spent fuel pit water, the STC will remain safely subcritical.

[IP3] Fuel Handling Accident

The probability of an FHA will not increase significantly due to the proposed changes because the individual fuel assemblies will be moved between the STC and the spent fuel pit racks and the STC and HI-TRAC will be moved in the same manner, using the same equipment, procedures, and other administrative controls (i.e. fuel move sheets) that are currently used.

The consequences of the existing fuel handling accident remain bounding because only IP3 fuel is moved in the IP3 spent fuel pit. The IP3 fuel assemblies to be transferred to IP2 will be cooled a minimum of 6 years. This compares with a cooling time of 84 hours used in the existing FHA radiological analysis. The 6-year cooling time results in a significant reduction in the radioactive source term available for release from a damaged fuel assembly compared to the source term considered in the design basis FHA radiological analysis. The consequences of the previously analyzed fuel assembly drop accident, therefore, continue to provide a bounding estimate of offsite dose for this accident.

[IP3] Loss of Spent Fuel Pool Cooling

The probability of a loss of spent fuel pit cooling remains the same because the proposed change does not alter the manner in which the IP3 spent fuel cooling loop is operated, designed or maintained.

The consequences of a loss of spent fuel pit cooling remains the same because the thermal design basis for the spent fuel pit cooling loop provides for all fuel pit rack locations to be filled at the end of a full core discharge and therefore the design basis heat load effectively includes any heat load associated with the assemblies within the STC.

[IP3] Natural Events

The natural events considered include the following accidents 1.) a seismic event, 2.) high winds, tornado and tornado missiles, 3.) flooding and 4) a lightning strike.

The probability of natural event will not increase due to the proposed changes because there are no elements of the proposed changes that influence the occurrence of any natural event.

The consequences of a natural event will not increase due to the proposed changes because the structural analyses design limits continue to be met. A lightning strike may cause ignition of the VCT fuel but this event is addressed under STC thermal accidents.

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 accident previously evaluated?

Response: No.

The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents.

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 amendment would modify the TS to incorporate the results of revised criticality, thermal and shield and dose analyses. The margin of safety required by 10 CFR 50.58(b)(4) remains unchanged. New criticality evaluations for both the STC [and the IP2 SFP] confirm that operation in accordance with the proposed amendment continues to meet the required subcriticality margins. The thermal analyses demonstrate that the postulated accidents (rupture of the HI-TRAC water jacket, 50-gallon transported fuel tank rupture and fire, simultaneous loss of water from the water jacket and HI-TRAC annulus, fuel misload, hypothetical tipover, and crane malfunction) continue to meet their acceptance criteria without a significant loss of safety margin. The shielding and dose analyses demonstrate that the shielding and radiation protection requirements continue to be met without a significant loss of safety margin.

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: Jeanne Cho, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: James G. Danna.

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

Date of amendment request: March 28, 2017. A publicly available version is in ADAMS under Accession No. ML17087A374.

Description of amendment request: The amendments would revise the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Technical Specifications (TSs) to change the low level of the refueling water tank (RWT) to reflect a needed increase in the required borated water volume and change the allowable value of the RWT level-low function.

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

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

   Response: No.

   The proposed amendment increases the required volume of water in the RWT to maintain the existing design requirements. The increase is

necessary due to an increase in the RWT Level - Low RAS [recirculation actuation signal] setpoint, which allows more water to stay in the tank following a LOCA [loss-of-coolant accident]. The modification to the allowable value of the RWT level-low (function 5a) resolves a non-conservative TS per the guidance of Administrative Letter 98-10 "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety."

The RWT is not an accident initiator. The RWT is required to supply adequate borated water to perform its mitigation function as assumed in the accident analyses. With the proposed increase in the minimum required water volume, the RWT maintains its design margin for supplying the required amount of borated water to the reactor core and the containment sump.

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 previously evaluated?

Response: No.

The proposed amendment increases the required volume of water in the RWT to maintain the existing design requirements. The increase is necessary due to an increase in the RWT Level - Low RAS setpoint, which allows more water to stay in the tank following a LOCA. The modification to the allowable value of the RWT level-low (function 5a) resolves a non-conservative TS per the guidance of Administrative Letter 98-10 "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety."

The proposed amendment does not impose any new or different requirements. The change does not alter assumptions made in the safety analyses. The proposed change is consistent with the safety analyses assumptions and current plant operating practice.

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

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

Response: No.

The proposed amendment increases the required volume of water in the RWT to maintain the existing design requirements. The increase is necessary due to an increase in the RWT Level - Low RAS setpoint,

> which allows more water to stay in the tank following a loss-of-coolant accident. The modification to the allowable value of the RWT level-low (function 5a) resolves a non-conservative TS per the guidance of Administrative Letter 98-10 "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety."
>
> The proposed amendment does not affect the design, operation, and testing methods for systems, structures and components specified in applicable codes and standards (or alternatives approved for use by the NRC). With the proposed increase in the minimum required water volume, the RWT maintains its design margin for supplying the required amount of borated water to the reactor core and the containment sump. The RWT will continue to meet all of its requirements as described in the plant licensing basis (including the Updated Final Safety Analysis Report and the TS Bases). Similarly, there is no impact to Safety Analysis acceptance criteria as described in the plant licensing basis.
>
> 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: 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, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: April 5, 2017. A publicly available version is in ADAMS under Accession No. ML17095A081.

Description of amendment request: The amendment would revise the Nine Mile Point Nuclear Station, Unit 2, Technical Specifications to allow greater flexibility in performing surveillance testing in Modes 1, 2, or 3 of emergency diesel generators and Class 1E batteries. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-283-A, Revision 3, "Modify Section 3.8 Mode Restriction Notes."

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 modify Mode restriction Notes to allow performance of the Surveillance in whole or in part to reestablish Emergency Diesel Generator (EDG) Operability, and to allow the crediting of unplanned events that satisfy the Surveillances. The EDGs and their associated emergency loads are accident mitigating features, and are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. To manage any increase in risk, the proposed changes require an assessment to verify that plant safety will be maintained or enhanced by performance of the Surveillance in the current prohibited Modes. The radiological consequences of an accident previously evaluated during the period that the EDG is being tested to reestablish operability are no different from the radiological consequences of an accident previously evaluated while the EDG is inoperable. As a result, the consequences of any accident previously evaluated are not increased.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any 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 involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change

to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis.

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 purpose of Surveillances is to verify that equipment is capable of performing its assumed safety function. The proposed changes will only allow the performance of the Surveillances to reestablish Operability, and the proposed changes may not be used to remove an EDG from service. In addition, the proposed changes will potentially shorten the time that an EDG is unavailable because testing to reestablish Operability can be performed without a plant shutdown. The proposed changes also require an assessment to verify that plant safety will be maintained or enhanced by performance of the Surveillance in the normally prohibited Modes.

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.

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 27, 2017. A publicly available version is in ADAMS under Accession No. ML17121A449.

Description of amendment request: The proposed amendments would revise Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for the permanent extension of the Type A integrated leak rate testing and Type C leak rate testing frequencies, and would also delete a one-time exception.

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 activity involves revision of the Quad Cities Nuclear Power Station (QCNPS) Technical Specification (TS) 5.5.12, Primary Containment Leakage Rate Testing Program, to allow the extension of the QCNPS, Units 1 and 2, Type A containment integrated leakage rate test interval to 15 years, and the extension of the Type C local leakage rate test interval to 75 months. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The existing Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions.

   The proposed extension does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident.

   The change in dose risk for changing the Type A Integrated Leak Rate Test (ILRT) interval from three-per-ten years to once-per-fifteen-years,

measured as an increase to the total integrated dose risk for all internal events accident sequences for QCNPS, is 1.0E-02 person-rem/yr (0.31%) using the Electric Power Research Institute (EPRI) guidance with the base case corrosion included. The change in dose risk drops to 2.7E-03 person-rem/yr (0.08%) when using the EPRI Expert Elicitation methodology. The values calculated per the EPRI guidance are all lower than the acceptance criteria of less than or equal to 1.0 person-rem/yr or less than 1.0% person-rem/yr defined in Section 1.3 of Attachment 3 to this LAR. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

As documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated January 1995, Types B and C tests have identified a very large percentage of containment leakage paths, and the percentage of containment leakage paths that are detected only by Type A testing is very small. The QCNPS, Units 1 and 2 Type A test history supports this conclusion.

The integrity of the containment is subject to two types of failure mechanisms that can be categorized as: (1) activity based, and, (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with American Society of Mechanical Engineers (ASME) Section XI, and TS requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed test interval extensions do not significantly increase the consequences of an accident previously evaluated.

The proposed amendment also deletes an exception previously granted in amendments 220 and 214 to allow one-time extensions of the ILRT test frequency for QCNPS, Units 1 and 2, respectively. This exception was for an activity that has already taken place; therefore, this deletion is solely an administrative action that does not result in any change in how QCNPS, Units 1 and 2 are operated.

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

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

Response: No.

The proposed amendment to TS 5.5.12, “Primary Containment Leakage Rate Testing Program,” involves the extension of the QCNPS, Units 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant’s ability to mitigate the consequences of an accident.

The proposed change does not involve a physical modification to the plant (i.e., no new or different type of equipment will be installed), nor does it alter the design, configuration, or change the manner in which the plant is operated or controlled beyond the standard functional capabilities of the equipment.

The proposed amendment also deletes an exception previously granted under TS Amendments 220 and 214 to allow the one-time extension of the ILRT test frequency for QCNPS, Units 1 and 2, respectively. This exception was for an activity that has already taken place; therefore, this deletion is solely an administrative action that does not result in any change in how the QCNPS units are operated.

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

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

Response: No.

The proposed amendment to TS 5.5.12 involves the extension of the QCNPS, Units 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained.

The proposed change involves the extension of the interval between Type A containment leak rate tests and Type C tests for QCNPS, Units 1 and 2. The proposed surveillance interval extension is bounded by the 15-year ILRT interval and the 75-month Type C test interval currently authorized within NEI 94-01, Revision 3-A. Industry experience supports the conclusion that Types B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI

> and TS serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Types A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A and Type C test intervals.
>
> The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for QCNPS, Units 1 and 2. This exception was for an activity that has taken place; therefore, the deletion is solely an administrative action and does not change how QCNPS is operated and maintained. Thus, there is no reduction in any margin of safety.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: David J. Wrona.

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

Date of amendment request: April 9, 2017. A publicly available version is in ADAMS under Accession No. ML17101A637.

Description of amendment request: The amendments would modify the Technical Specifications (TSs) to remove various reporting requirements. Specifically, the amendments

would remove the requirements to prepare various special reports, the Startup Report, and the Annual Report. In addition, the amendments would revise the TSs to remove the completion time for restoring spent fuel pool water level to address inoperability of one of the two parallel flow paths in the residual heat removal or safety injection headers for the Emergency Core Cooling Systems and to make other administrative changes, including updating plant staff and responsibilities and correcting a misspelling.

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 actions, surveillance requirements, and administrative controls associated with the proposed changes to the technical specifications (TS) are not initiators of any accidents previously evaluated, so the probability of accidents previously evaluated is unaffected by the proposed changes. The proposed changes do not alter the design, function, operation, or configuration of any plant structure, system, or component (SSC). The capability of any operable TS-required SSC to perform its specified safety function is not impacted by the proposed changes. As a result, the outcomes of accidents previously evaluated are unaffected. Therefore, the proposed changes do not result in 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 challenge the integrity or performance of any safety-related systems. No plant equipment is installed or removed, and the changes do not alter the design, physical configuration, or method of operation of any plant SSC. No physical changes are made to the plant, so no new causal mechanisms are introduced. Therefore, the proposed changes to the TS 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 ability of any operable SSC to perform its designated safety function is unaffected by the proposed changes. The proposed changes do not alter any safety analyses assumptions, safety limits, limiting safety system settings, or method of operating the plant. The changes do not adversely impact plant operating margins or the reliability of equipment credited in the safety analyses. 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: William S. Blair, Managing Attorney - Nuclear, Florida Power & Light Company, 700 Universe Blvd., MS LAW/JB, Juno Beach, FL 33408-0420.

NRC Branch Chief: Undine S. Shoop.

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

Date of amendment request: April 20, 2017. A publicly available version is in ADAMS under Accession No. ML17111A631.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) Section 3.1.2, "Reactivity Anomalies," with a change to the method of calculating core reactivity for the purpose of performing the reactivity anomaly surveillance. The proposed change would allow performance of the reactivity anomaly surveillance on a

comparison of monitored to predicted core reactivity. The reactivity anomaly verification is currently determined by a comparison of monitored versus predicted control rod density.

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 does not affect any plant systems, structures, or components designed for the prevention or mitigation of previously evaluated accidents. The proposed change would only modify 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 TS requirements such that, rather than performing the surveillance by comparing monitored to predicted control rod density, the surveillance is performed by a direct comparison of core k_{eff}. Present day on-line core monitoring systems, such as 3D MONICORE and ACUMEN, 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 change does not involve a significant increase in the probability or consequence of a previously evaluated accident

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 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 Anomalies TS will only provide a new, more efficient method of detecting an unexpected change in core reactivity.

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

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

   Response: No.

   The proposed change is to modify the method for performing the reactivity anomaly surveillance from a comparison of monitored to predicted control rod density to a comparison of monitored to predicted core 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 TS, only the method of performing the surveillance will be changed. Consequently, core reactivity assumptions made in safety analyses will continue to be adequately verified. The proposed change has no impact to 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: William Blair, P.O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: David J. Wrona.

NextEra Energy Point Beach, LLC, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant (Point Beach), Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: March 31, 2017. A publicly available version is in ADAMS under Accession No. ML17090A511.

Description of amendment request: The amendments would document a risk-informed resolution strategy to resolve low risk, legacy design code non-conformances associated with construction trusses in the containment buildings of Point Beach, Units 1 and 2. The proposed license amendment request (LAR) is a risk-informed licensing basis change. The proposed

change is acceptance of the final configuration of the construction trusses, including the attached containment spray piping and ventilation ductwork, and the containment liners/walls adjacent to the trusses, using a risk-informed resolution. Accordingly, the proposed change meets the criteria set forth in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and the generic guidance in RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

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 probability of an accident previously evaluated is not changed. The containment structures and the containment spray piping and ventilation ducts attached to the construction trusses are accident mitigation equipment. They are not accident initiators.

   The acceptance of the final configuration of Point Beach Units 1 and 2 results in a change in core damage frequency and large early release frequency that is within acceptance guidelines and does not involve a significant reduction in the margin of safety. Although failures are postulated in the PRA analysis, the engineering calculations in support of the LAR conclude that the construction trusses and the associated structures/components remain structurally sound in the event of a design basis seismic or thermal event and there is no adverse impact or change to any station SSC's [structure, system, and components] design function and there is no change to accident mitigation response.

   This change has no impact on station fire risk caused by a seismic event.

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

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

Response: No.

The proposed change does not install any new or different type of equipment in the plant. The proposed change does not create any new failure modes for existing equipment or any new limiting single failures. Engineering calculations conclude the construction trusses, equipment supported by the trusses, and containment liners remain capable of withstanding design basis seismic and thermal events and remain capable of performing their designated design functions. Additionally, the proposed change does not involve a change in the methods governing normal plant operation, and all safety functions will continue to perform as previously assumed in the accident analyses. Thus, the proposed change does not adversely affect the design function or operation of any structures, systems and components important to safety.

There are no new accidents identified associated with acceptance of the final modified configuration of Unit 1 and the current configuration of Unit 2.

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

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

Response: No.

The effects of the change, ΔCDF [core damage frequency] and ΔLERF, [large early release frequency] are within the acceptance guidelines shown in Figures 4 and 5 of Regulatory Guide 1.174. Consequently, the change does not result in a significant reduction in the margin of safety.

The containment structures and liners, construction trusses, and equipment supported by the trusses remain fully capable of performing their specified design functions as concluded by supporting engineering calculations.

Modifications associated with implementation of NFPA [National Fire Protection Association] 805 are planned that will provide protection of the reactor coolant system feed and bleed capability and result in additional safety margin.

The proposed change does not affect the margin of safety associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed change does not alter any safety analyses assumptions, safety limits, limiting safety system settings, or methods of operating the plant. The

> changes do not adversely impact the reliability of equipment credited in the safety analyses. The proposed change does not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition.
>
> The station will implement new seismic and thermal event limits to ensure the construction trusses and associated equipment are inspected and/or analyzed for any event exceeding elastic stress limits to determine their capability to withstand a subsequent design basis event prior to Unit restart.
>
> 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: William Blair, Managing Attorney - Nuclear, Florida Power & Light Company, P.O. Box 14000, 700 Universe Boulevard, Juno Beach, FL 33408-0420.

NRC Branch Chief: David J. Wrona.

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

Date of amendment request: April 27, 2017. A publicly available version is in ADAMS under Accession No. ML17118A049.

Description of amendment request: The requested amendments propose changes to combined license (COL) Appendix C (and plant-specific Tier 1) Table 2.7.2-2 to revise the minimum chilled water flow rates to the supply air handling units serving the Main Control Room and the Class 1E electrical rooms, and the unit coolers serving the normal residual heat removal system

and chemical and volume control system pump rooms. The proposed COL Appendix C (and plant-specific Design Control Document (Tier 1) changes require additional changes to corresponding Tier 2 component data information in Updated Final Safety Analysis Report (UFSAR) Chapter 9. Because this proposed change requires a departure from Tier 1 information in the Westinghouse Electric Company's AP1000 Design Control Document, the licensee also requested an exemption from the requirements of the Generic Design Control Document Tier 1 in accordance with 10 CFR 52.63(b)(1).

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

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

   Response: No.

   The proposed changes to COL Appendix C (and plant-specific Tier 1) Table 2.7.2-2, Updated Final Safety Analysis Report (UFSAR) Table 9.2.7-1, and associated UFSAR design information to identify the revised equipment parameters for the nuclear island nonradioactive ventilation system (VBS) air (VAS) unit coolers and reduced chilled water system (VWS) cooling coil flow rates do not adversely impact the plant response to any accidents which are previously evaluated. The function of the cooling coils to provide chilled water to the VBS AHUs and VAS unit coolers is not credited in the safety analysis.

   No safety-related structure, system, component (SSC) or function is adversely affected by this change. The VWS safety-related function of containment isolation is not affected by this change. The change does not involve an interface with any SSC accident initiator or initiating sequence of events, and thus, the probabilities of the accidents evaluated in the plant-specific UFSAR are not affected. The proposed changes do not involve a change to the predicted radiological releases due to postulated accident conditions, thus, the consequences of the accidents evaluated in the UFSAR are not affected. The proposed changes do not increase the probability or consequences of an accident previously evaluated as the VWS, VBS and VAS do not provide safety-related functions and the functions of each system to support required room environments are not changed.

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 COL Appendix C (and plant-specific Tier 1) Table 2.7.2-2, UFSAR Table 9.2.7-1, and associated UFSAR design information to identify the revised equipment parameters for VBS AHUs and VAS unit coolers and reduced VWS cooling coil flow rates do not affect any safety-related equipment, and do not add any new interfaces to safety-related SSCs. The VWS function to provide chilled water is not adversely impacted. The function of the VAS to provide ventilation and cooling to maintain the environment of the serviced areas within the design temperature range is not adversely impacted by this change. No system or design function or equipment qualification is affected by these changes as the change does not modify the operation of any SSCs. The changes do not introduce a new failure mode, malfunction or sequence of events that could affect safety or safety-related equipment. Revised equipment parameters, including the reduced cooling coil flow rates, do not adversely impact the function of associated components.

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 changes to COL Appendix C (and plant-specific Tier 1) Table 2.7.2-2, UFSAR Table 9.2.7-1, and associated UFSAR design information do not affect any other safety-related equipment or fission product barriers. The requested changes will not adversely affect compliance with any design code, function, design analysis, safety analysis input or result, or design/safety margin. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the requested changes as previously evaluated accidents are not impacted.

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

NRC Branch Chief: Jennifer Dixon-Herrity.

Tennessee Valley Authority, Docket Nos. 50-259, 50-260, 50-296, and 72-052, Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, and Independent Spent Fuel Storage Installation (ISFSI), Limestone County, Alabama

Tennessee Valley Authority, Docket Nos. 50-327, 50-328, and 72-034, Sequoyah Nuclear Plant (SQN), Units 1 and 2, and ISFSI, Hamilton County, Tennessee

Tennessee Valley Authority (TVA), Docket Nos. 50-390, 50-391, and 72-1048, Watts Bar Nuclear Plant (WBN), Units 1 and 2, and ISFSI, Rhea County, Tennessee

Date of amendment request: January 4, 2017. A publicly available version is in ADAMS under Accession No. ML17004A340.

Description of amendment request: The amendments would modify the Emergency Plans for BFN, Units 1, 2, and 3, and its ISFSI; SQN, Units 1 and 2, and its ISFSI; and WBN, Units 1 and 2, and its ISFSI, to adopt the Emergency Action Level (EAL) schemes based on Nuclear Energy Institute (NEI) 99-01, Revision 6, which has been endorsed by the NRC as documented in a letter dated March 28, 2013 (ADAMS Accession No. ML12346A463). The proposed changes to TVA's EAL schemes to adopt the guidance in NEI 99-01, Revision 6, do not reduce the capability to meet the emergency planning requirements established in 10 CFR 50.47 and 10 CFR Part 50, Appendix E. The proposed changes do not reduce the functionality, performance,

or capability of TVA's Emergency Response Organization (ERO) to respond in mitigating the consequences of accidents. The TVA ERO functions will continue to be performed as required.

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

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

   Response: No.

   The proposed changes to TVA's EAL schemes to adopt the NRC-endorsed guidance in NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," do not reduce the capability to meet the emergency planning requirements established in 10 CFR 50.47 and 10 CFR [Part] 50, Appendix E. The proposed changes do not reduce the functionality, performance, or capability of TVA's ERO to respond in mitigating the consequences of any design basis accident.

   The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facilities or the manner in which the plants are operated and maintained. The proposed change does not adversely affect the ability of structures, systems, and components (SSC) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptable limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposure.

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

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

   Response: No.

   The proposed changes to TVA's EAL schemes to adopt the NRC-endorsed guidance in NEI 99-01, Revision 6, do not involve any physical changes to plant systems or equipment. The proposed changes do not

involve the addition of any new plant equipment. The proposed changes will not alter the design configuration, or method of operation of plant equipment beyond its normal functional capabilities. All TVA ERO functions will continue to be performed as required. The proposed changes do not create any new credible failure mechanisms, malfunctions, or accident initiators.

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

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

Response: No.

The proposed changes to TVA's EAL schemes to adopt the NRC-endorsed guidance in NEI 99-01, Revision 6, do not alter or exceed a design basis or safety limit. 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. There are no changes to setpoints or environmental conditions of any SSC or the manner in which any SSC is operated. Margins of safety are unaffected by the proposed changes to adopt the NEI 99-01, Revision 6, EAL scheme guidance. The applicable requirements of 10 CFR 50.47 and 10 CFR [Part] 50, Appendix E will continue to be met.

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

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

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

NRC Branch Chief: Benjamin G. Beasley.

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 Inc., Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson), Darlington County, South Carolina

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

Date of amendment request: August 19, 2015, as supplemented by letters dated May 4, October 3, and November 17, 2016.

Brief description of amendments: The amendments revised the Robinson Technical Specification (TS) 5.6.5.b and the Harris TS 6.9.1.6.2 to adopt the methodology reports DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors"; DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design"; and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," for application specific to Robinson and Harris.

Date of issuance: May 18, 2017.

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

Amendment Nos.: 253 (Robinson) and 157 (Harris). A publicly available version is in ADAMS under Accession No. ML17102A923; documents related to these amendments are listed in the Safety Evaluations enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-23 and NPF-63: Amendments revised the Renewed Facility Operating Licenses and TSs.

Date of initial notice in *Federal Register*: February 2, 2016 (81 FR 5492). The supplemental letter dated May 4, 2016, provided additional information that expanded the scope of the application as originally noticed, and changed the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*. Accordingly, the NRC published a second proposed no significant hazards consideration determination in the

Federal Register on August 2, 2016 (81 FR 50746). This notice superseded the original notice in its entirety. The supplemental letters dated October 3 and November 17, 2016, provided additional information that clarified the application, did not expand the scope beyond the second notice, and did not change the NRC staff's proposed no significant hazards consideration determination as published in the *Federal Register.*

The Commission's related evaluations of the amendments are contained in the Safety Evaluations dated May 18, 2017.

No significant hazards consideration comments received: No.

Duke Energy Progress, LLC, Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendment request: December 21, 2015, as supplemented by letters dated June 29, July 13, August 15, November 1, November 17, 2016, and February 27, 2017.

Brief description of amendments: The amendments adopted the approved changes to Standard Technical Specifications for General Electric (BWR/4) [Boiling Water Reactor] Plants, NUREG-1433, Revision 4, to allow relocation of specific technical specification surveillance frequencies to a licensee-controlled program. The changes are described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b" (ADAMS Package Accession No. ML090850642), and are described in the Notice of Availability published in the *Federal Register* on July 6, 2009 (74 FR 31996).

Date of issuance: May 24, 2017.

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

Amendment Nos.: 276 (Unit 1) and 304 (Unit 2). A publicly available version is in ADAMS under Accession No. ML17096A129; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. DPR-71 and DPR-62: Amendments revised the Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: March 29, 2016 (81 FR 17504). The supplemental letters dated June 29, July 13, August 15, November 1, November 17, 2016, and February 27, 2017, 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 May 24, 2017.

No significant hazards consideration comments received: No.

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: May 26, 2016, as supplemented by letter dated December 19, 2016.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) by adding a new Administrative Controls section to establish, implement, and maintain a Diesel Fuel Oil Testing Program. It also relocated to this program the current TS surveillance requirements (SRs) for evaluating diesel fuel oil, along with the SRs for draining, sediment removal, and cleaning of each main fuel oil storage tank at least once every 10 years. In addition, the licensee took an exception to NRC Regulatory Guide 1.137, Revision 1, "Fuel-Oil

Systems for Standby Diesel Generators," to allow for the ability to perform sampling of new fuel oil offsite prior to its addition to the fuel oil storage tanks.

Date of issuance: May 24, 2017.

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

Amendment No.: 158. A publicly available version is in ADAMS under Accession No. ML17048A184; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

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

Date of initial notice in *Federal Register*: October 11, 2016 (81 FR 70178). The supplemental letter dated December 19, 2016, 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 May 24, 2017.

No significant hazards consideration comments received: No.

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 29, 2016, as supplemented by letter dated November 4, 2016.

Brief description of amendment: The amendment revised the Shearon Harris Nuclear Power Plant, Unit 1, Technical Specification (TS) 3/4.11.1.4, "Liquid Holdup Tanks"; TS 3/4.11.2.5, "Explosive Gas Mixture"; and TS 6.8.4.j, "Gas Storage Tank Radioactivity Monitoring Program." The amendment deleted TS Definition 1.16, "GASEOUS RADWASTE TREATMENT SYSTEM"; TS 3/4.11.1.4, "Liquid Holdup Tanks"; and TS 3/4.11.2.5, "Explosive Gas Mixture." The amendment relocated the deleted requirements for these TSs to licensee control under TS 6.8.4.j, "Gas Storage Tank Radioactivity Monitoring Program." The description for TS 6.8.4.j, "Gas Storage Tank Radioactivity Monitoring Program," was modified to include the controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The amendment relocated requirements associated with TS 3/4.11.1.4 and TS 3/4.11.2.5 to the licensee-controlled Plant Programs Procedure PLP-114, "Relocated Technical Specifications and Design Basis Requirements."

Date of issuance: May 25, 2017.

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

Amendment No.: 159. A publicly available version is in ADAMS under Accession No. ML17074A672; 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 Facility Operating License and TSs.

Date of initial notice in *Federal Register*: October 25, 2016 (81 FR 73433). The supplemental letter dated November 4, 2016, 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 May 25, 2017.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 11, 2016.

Brief description of amendment: The amendment approved adoption of NRC-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler TSTF-545, Revision 3, "TS [Technical Specification] Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015. Specifically, the amendment deleted Palisades Nuclear Plant TS 5.5.7, "Inservice Testing Program," and added a new defined term, "INSERVICE TESTING PROGRAM," to the TSs. All existing references to the "Inservice Testing Program," in the Palisades Nuclear Plant TS SRs are replaced with "INSERVICE TESTING PROGRAM" so that the SRs refer to the new definition in lieu of the deleted program.

Date of issuance: May 30, 2017.

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

Amendment No.: 262. A publicly available version is in ADAMS under Accession No. ML17082A465; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-20: Amendment revised the Renewed Facility Operating License and TSs.

Date of initial notice in *Federal Register*: August 30, 2016 (81 FR 59663).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 2017.

No significant hazards consideration comments received: No.

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

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

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

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. 50-220 and 50-410, Nine Mile Point Nuclear Station, Units 1 and 2, Oswego County, New York

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

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

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

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

Date of amendment request: July 26, 2016, as supplemented by letter dated October 6, 2016.

Brief description of amendments: The amendments revised the Inservice Testing Program requirements in each plant's technical specifications (TSs). The changes included deleting the current TS requirements for the Inservice Testing Program, adding a new defined term, "INSERVICE TESTING PROGRAM," to the TSs, and revising other TSs to reference this new defined term instead of the deleted program.

Date of issuance: May 26, 2017.

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

Amendment Nos.: 191,192,197,197, 320, 298, 212, 254, 247, 223, 209, 227, 161, 313, 317, 266, 261, 124, and 290. A publicly available version is in ADAMS under Accession No. ML17073A067. Documents related to these amendments are listed in the Safety Evaluations enclosed with the amendments.

Facility Operating License Nos.: NPF-72, NPF-77, NPF-37, NPF-66, DPR-53, DPR-69, NPF-62, DPR-19, DPR-25, NPF-11, NPF-18, DPR-63, NPF-69, DPR-44, DPR-56, DPR-29, DPR-30, DPR-18, and DPR-50. Amendments revised the Facility Operating Licenses and TSs.

Date of initial notice in *Federal Register*: November 8, 2016 (81 FR 78648).

The Commission's related evaluations of the amendments are contained in Safety Evaluations dated May 26, 2017.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit 2, Beaver County, Pennsylvania

Date of amendment request: June 24, 2016, as supplemented by letters dated September 13, 2016; December 15, 2016; and March 16, 2017.

Brief description of amendment: The amendment modified the Renewed Facility Operating License to reflect the direct transfer of Toledo Edison Company's 18.26 percent leased interest in Beaver Valley Power Station, Unit 2, and Ohio Edison Company's 21.66 percent leased interest in Beaver Valley Power Station, Unit 2, from FirstEnergy Nuclear Operating Company to FirstEnergy Nuclear Generation, LLC.

Date of issuance: May 30, 2017.

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

Amendment No.: 187. A publicly available version is in ADAMS under Accession No. ML17115A123.

Renewed Facility Operating License No. NPF-73: Amendment revised the Renewed Facility Operating License.

Date of initial notice in *Federal Register*: January 23, 2017 (82 FR 7880). The supplemental letter dated March 16, 2017, 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 April 14, 2017.

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: July 21, 2016, as supplemented by letter dated September 26, 2016.

Brief description of amendments: The amendments revised the Donald C. Cook Nuclear Plant, Units 1 and 2, Technical Specification (TS) Surveillance Requirements (SRs), consistent with the NRC-approved Technical Specifications Task Force (TSTF) Traveler, TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing." Specifically, the change revised the TSs to eliminate Section 5.5.6, "Inservice Testing Program." A new defined term, "INSERVICE TESTING PROGRAM," was added to the TS Definitions section. TS SRs that previously referred to the Inservice Testing Program from Section 5.5.6 were revised to refer to the new defined term, "INSERVICE TESTING PROGRAM."

Date of issuance: May 24, 2017.

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

Amendment Nos.: 335 (Unit 1) and 317 (Unit 2). A publicly available version is in ADAMS under Accession No. ML17103A106; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Renewed Facility Operating Licenses and TSs.

Date of initial notice in *Federal Register*: September 27, 2016 (81 FR 66307). The supplemental letter dated September 26, 2016, 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 May 24, 2017.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 18, 2016, as supplemented by letters dated February 10, 2017; March 1, 2017; and March 10, 2017.

Brief description of amendments: The amendments revised Technical Specification 3.14 "Circulating and Service Water Systems," to extend the Allowed Outage Time for only one operable Service Water flow path to the Changing Pump Service Water subsystem and to the Main Control Room/Emergency Switchgear Room air conditioning subsystem.

Date of issuance: May 31, 2017.

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

Amendment Nos.: 289 (Unit 1) and 289 (Unit 2). A publicly available version is in ADAMS under Accession No. ML17100A253; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. NPF-4 and NPF-7: Amendments revised the Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: October 25, 2016 (81 FR 73443). The supplemental letters dated February 10, 2017; March 1, 2017; and March 10, 2017, 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 May 31, 2017.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 6th day of June 2017.

For the Nuclear Regulatory Commission.

/RA/

Eric J. Benner, Deputy Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.