[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2019-0238]

Biweekly Notice

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from November 5, 2019 to November 18, 2019. The last biweekly notice was published on November 19, 2019.

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

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

- **Federal Rulemaking Web Site:** Go to https://www.regulations.gov and search for Docket ID **NRC-2019-0238**. Address questions about NRC dockets IDs in Regulations.gov to Jennifer Borges; telephone: 301-287-9127; e-mail: Jennifer.Borges@nrc.gov. For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.

- **Mail comments to:** Office of Administration, Mail Stop: TWFN-7-A60M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff.

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: Paula Blechman, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone: 301-415-2242, email: Paula.Blechman@nrc.gov

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID **NRC-2019-0238**, 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 https://www.regulations.gov and search for Docket ID **NRC-2019-0238**.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document.

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

B. Submitting Comments

Please include Docket ID **NRC-2019-0238**, 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 https://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. Background

Pursuant to Section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the 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.

III. 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 section 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 https://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 to be made a party to the proceeding; (3) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the petitioner's interest.

In accordance with 10 CFR 2.309(f), the petition must also set forth the specific contentions which the petitioner seeks to have litigated in the proceeding. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner must provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue. The petition must include sufficient information to show that a genuine dispute exists with the applicant or licensee on a material issue of law or fact. Contentions must be limited to matters within the scope of the proceeding. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to satisfy the requirements at 10 CFR 2.309(f) with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene. Parties have the opportunity to participate fully in the conduct of the hearing with respect to resolution of that party's admitted contentions, including the opportunity to present evidence, consistent with the NRC's regulations, policies, and procedures.

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

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to establish when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, then any hearing held would take place before the issuance of the amendment unless the Commission finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR part 2.

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

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

B. Electronic Submissions (E-Filing)

All documents filed in NRC adjudicatory proceedings, including a request for hearing and petition for leave to intervene (petition), any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities that request to participate under 10 CFR 2.315(c), must be filed in accordance with the NRC’s E-Filing rule (72 FR 49139; August 28, 2007, as amended at 77 FR 46562; August 3, 2012). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Detailed guidance on making electronic submissions may be found in the Guidance for Electronic Submissions to the NRC and on the NRC Web site at https://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 https://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 https://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 https://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.

DTE Electric Company, Docket No. 50-341, Fermi 2, Monroe County, Michigan

Date of amendment request: September 5, 2019. A publicly-available version is in ADAMS under Accession No. ML19248C571.

Description of amendment request: The amendment would revise the Fermi 2 Technical Specification (TS) 2.1.1, "Reactor Core SLs [safety limits]," reactor steam dome pressure from 785 psig [pounds per square inch gauge] to 686 psig and TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 1.b, "Main Steam Line Pressure – Low," isolation function allowable value from 736 psig to 801 psig.

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

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

   Response: No.

   The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because decreasing the reactor steam dome pressure in TS Safety Limits 2.1.1.1 and 2.1.1.2 for reactor thermal power ranges and increasing the trip set point and allowable value for main steam line low pressure isolation effectively expands the validity range for GEXL critical power correlation and the calculation of minimum critical power ratio. The critical power ratio rises during the pressure reduction following the scram that terminates the PRFO [pressure regulator failure – Open] transient. The reduction in reactor steam dome pressure value in the SL and the increase in trip set point and the reactor steam dome pressure value in the SL and the increase in the trip set point and the allowable value for the main steam line low pressure isolation provides adequate margin to accommodate the pressure reduction during the PRFO transient within the revised TS limit.

   The proposed changes do not alter the use of the analytical methods used to determine the safety limits that have been previously reviewed and approved by the NRC. The proposed changes are in accordance with an NRC approved critical power correlation methodology and do not adversely affect accident initiators or precursors.

The proposed changes do not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the applicable acceptance limits. The proposed changes are consistent with the safety analysis and resultant consequences.

Based on the above, DTE has concluded that the proposed change will 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 previously evaluated?

   Response: No.

   The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed reduction in the reactor dome steam pressure value in the safety limit in conjunction with the increase in the trip setpoint and the allowable value for the main steam line low pressure isolation reflects a wider range of applicability for the GEXL critical power correlation which is approved by the NRC.

   In addition, no new failure modes are being introduced. There are no changes in the method by which any plant systems perform a safety function. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

   The proposed changes do not introduce any new accident precursors, nor do they involve any changes in the methods governing normal plant operation. The proposed changes do not alter the outcome of the safety analysis.

   Based on the above, DTE has concluded that the proposed TS 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 margin of safety is established through the design of the plant structures, systems, and components, and through the

> parameters for safe operation and setpoints for actuation of equipment relied upon to respond to transients and design basis accidents. Evaluation of the 10 CFR part 21 condition by General Electric determined that since the Minimum Critical Power Ratio improves during the PRFO transient, there is no decrease in the safety margin and therefore there is not a threat to fuel cladding integrity. The proposed change in reactor steam dome pressure limits supports the current safety margin, which protects the fuel cladding integrity during a depressurization transient, but does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change does not alter the behavior of plant equipment, which remains unchanged. By raising the MSL LPIS AV [main steamline, low-pressure injection system, allowable value] in conjunction with lowering the Reactor Steam Dome Pressure SL, there is an increase in margin which increases protection of the MCPR [maximum critical power ratio].
>
> The proposed change to Reactor Core SLs 2.1.1.1 and 2.1.1.2 is consistent with and within the capabilities of the applicable NRC approved critical power correlation for the fuel designs in use at Fermi 2. The proposed change does not alter the manner in which the SLs are determined. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.
>
> The reduction in value of the reactor steam dome pressure safety limit and the increase in the trip setpoint and allowable value for main steam line low pressure isolation provides adequate margin to accommodate the pressure reduction during the PRFO transient within the revised TS limit.
>
> Based on the above, DTE has concluded that the proposed TS change does not involve a significant reduction in the margin of safety.

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

<u>Attorney for licensee</u>: Jon P. Christinidis, DTE Energy, 688 WCB, One Energy Plaza, Detroit, MI 48226.

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

Duke Energy Progress, LLC, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: July 29, 2019. A publicly-available version is in ADAMS under Accession No. ML19210D020.

Description of amendment request: The amendment would revise H. B. Robinson Steam Electric Plant, Unit No. 2, Technical Specification (TS) 3.7.3 regarding main feedwater isolation valves, main feedwater regulation valves, and bypass valves, by making the TS applicable to three additional feedwater bypass valves. The amendment would also revise the condition and completion time associated with the feedwater bypass valves.

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 amendment does not modify the feedwater system, nor does it make any physical or operational changes to the facility. The new non-safety BVs [bypass valves] are being installed under 10 CFR 50.59 to provide a backup isolation function to the existing safety grade BVs, consistent with NUREG-0138 and Section 6.2.1.4 of the NRC's Standard Review Plan. The new BVs will receive the same Engineered Safety Features signals to close and they will be subject to the same testing as the existing safety grade BVs. The proposed change has no impact on the containment or accident analyses. Inclusion of the new BVs within the scope of TS 3.7.3 subjects them to the same TS LCO [limiting condition for operation] and Surveillance Requirements as the existing BVs and allows them to be credited as backups to the existing BVs.

Extending the Completion Time of TS 3.7.3, Required Action C.1 from 8 hours to 72 hours is not an accident initiator and thus does not change the probability that an accident will occur; however, it could potentially affect the consequences of an accident if the accident occurred during the extended unavailability of an inoperable BV. The new BVs provide redundant isolation in the feedwater bypass flow paths. This represents a safety improvement over the original single BV (per flow path) design. The proposed increase in time an inoperable BV is allowed to remain open/unisolated is small and the probability of an event requiring isolation of the feedwater flow path occurring during this period, coincident with a failure of the redundant BV in that flow path, is low.

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

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

   Response: No.

   The proposed amendment does not modify the feedwater system, nor does it make any physical or operational changes to the facility. Neither the inclusion of the new BVs in TS 3.7.3 nor the extension of the Completion Time for TS 3.7.3 Required Action C.1 results in any new failure modes or affects. The new non-safety BVs are being installed under 10 CFR 50.59 to provide a backup isolation function to the existing safety grade BVs. Closure of the BVs is required to mitigate the consequences of steam line and feedwater line break events. The proposed changes allow for the new BVs to be credited in plant analyses for the isolation feedwater flow in the event of a failure of the existing BVs to close.

   Therefore, the possibility of a new or different kind of accident from any kind of accident previously evaluated is not created.

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

   Response: No.

   The proposed amendment does not involve: 1) a physical alteration of the plant, 2) a change to any set points for parameters associated with protection or mitigation actions nor 3) any impact on the fission product barriers or parameters associated with licensed safety limits. The new BVs are being

> installed under 10 CFR 50.59 to provide a backup isolation function to the existing BVs. There are no changes to either the containment analysis or to the analysis for any design basis event.
>
> Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Kathryn B. Nolan, Deputy General Counsel, Duke Energy Corporation, 550 South Tryon Street, DEC45A, Charlotte, NC 28202.

NRC Branch Chief: Undine Shoop.

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit 2 (ANO-2), Pope County, Arkansas

Date of amendment request: August 29, 2019. A publicly-available version is in ADAMS under Accession No. ML19241A264.

Description of amendment request: The proposed amendment would modify multiple Technical Specifications (TSs) for ANO-2 to address non-conservative TSs associated with the movement of fuel assemblies. This proposed change is necessary due to the previous adoption of the Alternate Source Terms, which included an update to the ANO-2 fuel handling accident (FHA) analysis. This update created a new requirement to address the movement of new (unirradiated) fuel assemblies over irradiated fuel assemblies. The proposed amendment would also adopt certain changes to gain greater consistency with NUREG-1432, Revision 4, "Standard Technical Specifications, Combustion Engineering Plants." The changes necessary to support the revised FHA affect similar TSs associated with Technical Specifications Task Force (TSTF) Standard Technical Specification Change Travelers TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations”; TSTF-272, Revision 1, “Refueling Boron Concentration Clarification”; TSTF-286, Revision 2, “Operations Involving Positive Reactivity Additions”; TSTF 471, Revision 1, “Eliminate Use of Term Core Alterations in ACTIONS and Notes”; and TSTF-571-1, Revision 0, “Revise Actions for Inoperable Source Range Neutron Flux Monitor.” Therefore, the licensee proposes to adopt these TSTFs in conjunction with changes necessary to support the revised FHA analysis. Additionally, the proposed amendment would incorporate specified administrative and editorial changes associated with the TS pages affected by the aforementioned proposed changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. Each of the six items described above is addressed under each of the three standards, 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.

   Updated FHA [Analysis]

   TS changes associated with the updated FHA analysis ensure the initial assumptions of the FHA are maintained and, therefore, act to minimize the consequences of an accident by ensuring TS required features are operable during the movement of fuel assemblies. The updated FHA analysis was previously accepted by the NRC during adoption of Alternate Source Terms (AST) for ANO-2. The probability of a fuel assembly drop (or any load drop) is unchanged by the updated FHA analysis. Therefore, the updated FHA analysis does not involve a significant increase in the probability of an accident previously evaluated.

Entergy has reviewed station procedures and controls in order to verify that no other loads, other than a new or irradiated fuel assembly, need be addressed with regard to an FHA (i.e., no other known load carried over irradiated fuel assemblies exists which would not be bounded by the fuel drop analysis or be expected to cause fuel damage if dropped). The proposed TS changes ensure required systems are operable during operations that could lead to an FHA. As previously approved by the NRC via the adoption of AST for ANO-2, the updated FHA analysis adequately bounds Control Room and offsite dose within federal limitations. Based on the above, the proposed FHA-related changes to the TSs do not result in a significant increase in the consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TSTF-51 and TSTF 471

The design basis accident (DBA) assumed for ANO-2 related to the proposed changes is the FHA. The boron dilution event is evaluated in the ANO-2 Safety Analysis Report (SAR), but [is] considered an unlikely event due to the time available for operator detection and response, along with prevalent administrative controls. A loss of Shutdown Cooling (SDC) event has little relationship to and minimal impact with regard to an FHA. TSTF-51 and TSTF-471 replace the use of the previously defined "core alterations" term with requirements associated with the movement of fuel assemblies, since the drop of a fuel assembly is the only event that could reasonably lead to an FHA or a significant challenge to the plant.

In addition, TSTF-51 reduces restrictions following sufficient radioactive decay of fuel assemblies since the offsite dose consequences of an FHA following this decay period (100 hours for ANO-2) would remain within 10 CFR 50.67 limits. Note that this allowance is not adopted for TS Control Room ventilation or radiation monitoring systems (associated with meeting 10 CFR 50, appendix A, General Design Criteria (GDC) 19).

The removal of references to "core alterations" in favor of restrictions associated with the movement of fuel assemblies eliminates current restrictions associated with the manipulation of other core components (i.e., sources or reactivity control components within the core) since such manipulation cannot result in an FHA, boron dilution event, or loss of SDC. In addition, manipulation of these other components cannot present a significant challenge to shutdown margin (SDM) because the TS required RCS boron concentration for Mode 6 operation provides substantial margin to criticality.

Changes associated with TSTF-51 and TSTF-471, as adopted, do not modify limitations in such a way that the consequences of an FHA would be greater than that assumed in the updated FHA analysis (i.e., 10 CFR 50.67 and GDC 19 limitations are not exceeded following an FHA).

Based on the above, the proposed changes associated with the adoption of TSTF-51 and TSTF-471 do not result in a significant increase in the probability or consequences of an accident previously evaluated.

TSTF-272

Changes associated with TSTF-272 place additional restrictions on Mode 6 operations by ensuring the boron concentration of the water in the refueling canal meets the same TS limits required for the Reactor Coolant System (RCS) when the RCS is in direct hydraulic communication with the refueling canal (i.e., reactor vessel head removed and refueling canal filled). These changes are unrelated to any accident initiator and further prohibit any challenge to the fuel in the reactor vessel by ensure sufficient boron concentration is maintained during Mode 6 operations. Therefore, these changes do not result in a significant increase in the probability or consequences of an accident previously evaluated.

TSTF-286

Changes associated with TSTF-286 permit operator control of RCS inventory and temperature when certain TS requirements are not met, provide[d] the overall required SDM of the RCS is maintained. The activities that involve inventory makeup from sources with boron concentrations less than the current RCS concentration (i.e., boron dilution) need not be precluded in the TSs provided the required SDM is maintained for the worst-case overall effect on the core. Note that an unexpected boron dilution event is considered unlikely for ANO-2 due to the significant period of time for operator detection and response before SDM would be significantly challenged (reference ANO-2 Safety Analysis Report Section 15.1.4.3). In addition, while a boron dilution event is evaluated in the accident analysis, the only “accident” assumed for ANO-2 during Mode 6 operations is the FHA. Permitting RCS inventory and temperature adjustments is unrelated to any assumptions associated with an FHA. Therefore, these changes do not result in a significant increase in the probability an accident (or a boron dilution event) previously evaluated. Because an unexpected boron dilution event provides sufficient opportunity for detection and recovery, the proposed changes associated with TSTF-286 likewise do not result in a significant increase in the consequences of an accident (or boron dilution event) previously evaluated.

TSTF-571-T

The proposed change revises the Actions for inoperable source range neutron flux monitors to prohibit the movement of fuel assemblies, sources, and reactivity control components when [a] monitor is inoperable. The Actions taken when a monitor is inoperable are not initiators to any accident previously evaluated. The monitors are not credited to mitigate any previously evaluated accident. The proposed change restricts the licensee's actions while a monitor is inoperable beyond the current requirements. Therefore, the consequences of an accident previously evaluated are not significantly increased.

Administrative/Editorial/Miscellaneous Changes

Enhancements and administrative changes proposed for TSs affected by the previously discussed updated FHA or changes associated with increasing consistency with the ITS [improved technical specifications] are unrelated to any accident initiator. Administrative changes likewise cannot impact the consequences of any accident previously evaluated.

The following is a listing of other changes proposed in this amendment request which modify the TSs (not considered within the editorial/administrative realm).

- A new Note 3 is proposed that clarifies the original intent of the TS requirements for radiation monitoring and automatic isolation of the Containment Purge system. As written, the TS would require the radiation monitoring and isolation capability to remain operable even when the Containment Purge system is secured. The addition of Note 3 specifies that operability is required only during 1) Containment Purge operations, or 2) ongoing Containment Building continuous ventilation operations when moving recently irradiated fuel assemblies or moving new fuel assemblies over irradiated fuel assemblies in the Containment Building, consistent with the updated FHA and TSTF-51. Other associated enhancements are made to the Containment Purge requirements in support of the above changes or to provide additional clarification.

- The phrase "elevation corresponding to the" top of irradiated fuel is added to the Limiting Condition for Operation (LCO) of TS 3.9.9, "Water Level - Reactor

Vessel." This ensures that proper water level is established prior to initiating refueling of the reactor core following a defueled condition.

- The movement of fuel "within the reactor vessel" contained in the Applicability and Action of TS 3.9.9 is revised to "within the Containment Building." This reference is also added to the Surveillance Requirement. The required water level should be met even when fuel is being moved in other areas of the refueling canal, not just in the reactor vessel. In addition, the phrase "while in Mode 6" is deleted from the Applicability since fuel assemblies cannot physically be removed from the reactor until Mode 6 has been achieved.

Enhancements associated with the Containment Purge system radiation instrumentation ensure Surveillance testing is performed when the system is in service, regardless if an actual Purge is taking place. In addition, the proposed changes ensure appropriate testing is performed prior to placing the system in service each refueling outage. The proposed changes are neutral or more restrictive and, therefore, cannot increase the consequences of an accident previously evaluated.

Clarifications to limitations on refueling water level and the location of fuel assemblies are more restrictive changes, ensuring proper controls have been established before activities are commenced. No impact to the consequences of any accident result from these changes. The changes to these TSs, in addition to the aforementioned changes to Containment Purge requirements, do not increase the probability of an accident occurring.

Based on the above, the proposed changes do not represent 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.

   Updated FHA [Analysis]

   TS changes associated with the updated FHA [analysis] involve no physical changes to the plant. These changes act to ensure required structures, systems, and components (SSCs) are operable when moving irradiated fuel assemblies or new fuel assemblies over irradiated fuel assemblies to limit any Control

Room or offsite dose consequences to within acceptable limits. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TSTF-51 and TSTF 471

TS changes associated with ITS improvements related to these TSTFs involve no physical changes to the plant. The removal of references to "core alterations" in favor of restrictions associated with the movement of fuel assemblies eliminates current restrictions associated with the manipulation of other core components (i.e., sources or reactivity control components within the core). Such manipulations cannot result in an FHA, boron dilution event, or loss of SDC. In addition, such manipulations cannot result in an appreciable change in core reactivity due to the high RCS boron concentration required during refueling operations by the TSs. TSTF-51 changes associated with a reduction in restrictions following sufficient radioactive decay of fuel assemblies are not considered accident precursors. The proposed changes do not introduce a new accident initiator, accident precursor, or accident-related malfunction mechanism. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TSTF-272

Changes associated with TSTF-272 place additional restrictions on Mode 6 operations by ensuring the boron concentration of the water in the refueling canal meets the same TS limits required for the RCS when the RCS is in direct hydraulic communication with the refueling canal (i.e., reactor vessel head removed and refueling canal filled). These changes are unrelated to any accident initiator and further prohibit any challenge to the fuel in the reactor vessel by [ensuring] sufficient boron concentration is maintained during Mode 6 operations. The proposed changes do not introduce a new accident initiator, accident precursor, or accident-related malfunction mechanism. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TSTF-286

Changes associated with TSTF-286 permit operator control of RCS inventory and temperature when certain TS requirements are not met, provide[d] the overall required SDM of the RCS is maintained. No physical plant changes are related to these TS changes. The only accident or event that could be affected by this change is the boron dilution event, which has been previously evaluated. The proposed changes do not introduce a new accident initiator, accident precursor, or accident-related malfunction mechanism. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TSTF-571-T

The proposed change revises the Actions for inoperable source range neutron flux monitors to prohibit the movement of fuel assemblies, sources, and reactivity control components when a monitor is inoperable. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). No credible new failure mechanisms, malfunctions, or accident initiators that would have been considered a design basis accident in the ANO-2 Safety Analysis Report (SAR) are created.

Administrative/Editorial/Miscellaneous Changes

Enhancements and administrative changes proposed for TSs affected by the above updated FHA or ITS improvements are unrelated to any accident initiator and involve no physical changes to the plant.

Enhancements associated with the Containment Purge system radiation instrumentation ensure Surveillance testing is performed when the system is in service, regardless if an actual Purge is taking place. In addition, the proposed changes ensure appropriate testing is performed prior to placing the system in service each refueling outage. Clarifications to limitations on refueling water level and the location of fuel assemblies are more restrictive changes, ensuring proper controls have been established before activities are commenced.

The proposed changes do not introduce a new accident initiator, accident precursor, or accident-related malfunction mechanism. Based on the above, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Updated FHA [Analysis]

TS changes associated with the updated FHA [analysis] act to ensure required SSCs are operable when moving irradiated fuel assemblies or new fuel assemblies over irradiated fuel assemblies to limit any Control Room or offsite dose consequences to within acceptable limits. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

TSTF-51 and TSTF 471

The removal of references to "core alterations" in favor of restrictions associated with the movement of fuel assemblies eliminates current restrictions associated with the manipulation of other core components (i.e., sources or reactivity control components within the core). Such manipulations cannot result in an FHA, boron dilution event, or loss of SDC. In addition, such manipulations cannot result in an appreciable change in core reactivity due to the high RCS boron concentration required during refueling operations by the TSs. TSTF-51 also reduces restrictions following sufficient radioactive decay of fuel assemblies since the consequence of an FHA following this decay period would remain within 10 CFR 50.67 limits. Note that this allowance is not adopted for Control Room ventilation or radiation monitoring systems (governed under GDC 19). Changes associated with TSTF-51 and TSTF-471, as adopted, do not modify limitations in such a way that the consequences of an FHA would be greater than that assumed in the FHA analysis (i.e., 10 CFR 50.67 and GDC 19 limitations are not exceeded following an FHA). Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

TSTF-272

Changes associated with TSTF-272 place additional restrictions on Mode 6 operations by ensuring the boron concentration of the water in the refueling canal meets the same TS limits required for the RCS when the RCS is in direct hydraulic communication with the refueling canal (i.e., reactor vessel head removed and refueling canal filled). These changes are more restrictive than the current TS and, therefore, do not involve a significant reduction in a margin of safety.

TSTF-286

Changes associated with TSTF-286 permit operator control of RCS inventory and temperature when certain TS requirements are not met, provide the overall required SDM of the RCS is maintained. The only accident or event that could be affected by this change is the boron dilution event which has been previously evaluated. While the margin between existing boron concentration and that required to meet SDM requirements may be reduced, margin is gained by permitting operators to take corrective action to maintain RCS inventory and temperature within limits during periods when such operations are otherwise prohibited. While not quantifiable, the changes associated with TSTF-286 have a general balanced effect in relation to the margin of safety. Because an unexpected boron dilution event provides sufficient opportunity for detection and recovery, the proposed changes associated with TSTF-286 do not involve a significant reduction in a margin of safety.

TSTF-571-T

The proposed change revises the Actions for inoperable source range neutron flux monitors to prohibit the movement of fuel assemblies, sources, and reactivity control components when a monitor is inoperable. No safety limits are affected. No Limiting Conditions for Operation or Surveillance limits are affected. The design, operation, surveillance methods, and acceptance criteria specified in applicable codes and standards (or alternatives approved for use by the NRC) continue to be met as described in the plants' [plant's] licensing basis. The proposed change does not adversely affect existing plant safety margins, or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Administrative/Editorial/Miscellaneous Changes

Enhancements and administrative changes proposed for TSs affected by the above updated FHA or ITS improvements are unrelated to any accident initiator or mitigation strategy. Enhancements associated with the Containment Purge system radiation instrumentation ensure Surveillance testing is performed when the system is in service, regardless if an actual Purge is taking place. In addition, the proposed changes ensure appropriate testing is performed prior to placing the system in service each refueling outage. Clarifications to limitations on refueling water level and the location of fuel assemblies are more restrictive changes, ensuring proper controls have been established before activities are commenced. Based on the above, these proposed changes do not involve a significant reduction in a margin of safety.

Therefore, the proposed changes contained within this amendment request 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: Ms. Anna Vinson Jones, Senior Counsel, Entergy Services, LLC, 101 Constitution Avenue, NW, Suite 200 East, Washington, DC 20001.

NRC Branch Chief: Jennifer L. Dixon-Herrity.

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

Date of amendment request: September 12, 2019. A publicly available version is in ADAMS under Accession No. ML19255D988.

Description of amendment request: The amendment would revise Technical Specifications related to primary containment hydrodynamic loads.

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 revise operating limits for containment systems during normal operation that provide the initial conditions at which containment performance to mitigate loss-of-coolant accidents is evaluated. The affected parameters are unrelated to the Reactor Coolant Pressure Boundary or reactivity control systems and therefore are unrelated to accident initiation or probability of occurrence.

Analysis has demonstrated that the containment will continue to operate within design limits in the event of an accident. Therefore, the consequences of an accident are not significantly affected by the proposed change.

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 do not alter the protection system design, create new failure modes, or change any modes of operation. The proposed changes do not involve a physical alteration of the plant; and no new or different kind of equipment will be installed. Consequently, there are no new initiators that could result in a new or different kind of accident.

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

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

   Response: No.

   The proposed changes will eliminate the 1.7 psi [pounds per square inch] differential pressure requirement between the drywell and wetwell, raise the maximum torus water level to 14.25 ft, and raise the HPCI [high pressure coolant injection] "Suppression Pool Water Level - High" Allowable Value to ≤ [less than or equal to] 14.75 ft. Technical Report "13-0541-TR-002" evaluated use of these operating parameters and determined that all structural elements continue to meet code requirements with adequate margin. Other design aspects such as Emergency Core Cooling System Pump Net Positive Suction Head, Equipment Qualification, and accident radiological dose impacted by the proposed changes were also evaluated and found to have negligible to no impact.

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

NRC Branch Chief: James G. Danna.

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

Date of amendment request: August 19, 2019. A publicly-available version is in ADAMS under Accession No. ML19238A065.

Description of amendment request: The proposed amendment would revise CNS Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for an exception to certain leak rate testing interval requirements of the program. Specifically, the proposed amendment would permit the 10 CFR part 50, appendix J, Option B leak testing of Type C residual heat removal system heat exchanger relief valves and their associated Type B testable discharge flange tests be performed at the same frequency as the visual examination, seat leakage testing, and set pressure testing performed for these valves under the requirements of the Inservice Testing Program per 10 CFR 50.55a(f).

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

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

   Response: No.

   The proposed change allows certain leak testing intervals required by the CNS primary containment leakage rate testing program to be aligned with certain testing intervals required by the Inservice Testing Program under 10 CFR50.55a(f). The containment function is solely to mitigate the consequences of an accident. No design basis accident is initiated by a failure of the containment leakage mitigation function. Aligning the testing interval requirements of the two programs does not create any adverse interactions with other systems that could result in initiation of a design basis accident. Continued containment integrity is assured by the established programs for local leakage rate testing and inservice testing which are unaffected by the proposed change.

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

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

   Response: No.

   The proposed change allows certain leak testing intervals required by the CNS primary containment leakage rate testing program to be aligned with certain testing intervals required by the Inservice Testing Program under 10 CFR 50.55a(f). This proposed change does not modify existing structures, systems, or components (SSC) of the plant, and it does not introduce new SSC's. The plant will continue to be operated in the same manner. Thus, it does not affect the design function or operation of SSC's involved, and it does not introduce a new accident initiator.

   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 change allows certain leak testing intervals required by the CNS primary containment leakage rate testing program to be aligned with certain testing intervals required by the Inservice

> Testing Program under 10 CFR 50.55a(f). The proposed alignment of testing intervals will not result in a change to the design or operation of any plant SSC used to shutdown the plant, initiate Emergency Core Cooling systems, or isolate the ability of CNS to mitigate any accident or transient. There is no impact on safety limits or limiting safety system settings. The change does not affect any plant safety parameters or setpoints.
>
> 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.

<u>Attorney for licensee</u>: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602-0499.

<u>NRC Branch Chief</u>: Jennifer Dixon-Herrity.

<u>NextEra Energy Duane Arnold (NEDA), LLC, Docket No. 50-331, Duane Arnold Energy Center (DAEC), Linn County, Iowa</u>

<u>Date of amendment request</u>: June 20, 2019, as supplemented by letters dated September 12, 2019, and November 4, 2019. Publicly-available versions are in ADAMS under Accession Nos. ML19176A356, ML19261A141, and ML19308A085, respectively.

<u>Description of amendment request</u>: The NRC staff has previously made a proposed determination that the amendment request dated June 20, 2019, involves no significant hazards consideration (84 FR 45544; August 29, 2019). Subsequently, the licensee provided additional information that expanded the scope of the amendment request as originally noticed. In the supplemental letter dated September 12, 2019, the licensee provided no significant hazards consideration for the supplemental changes only. This

notice combines the two no significant hazards considerations provided by the licensee. Accordingly, this notice supersedes the previous notice in its entirety.

By letter dated June 20, 2019, NEDA submitted a request for an amendment to the operating license (OL) and technical specifications (TSs) for the DAEC. The submittal requested revisions to the OL and TSs consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The revised TSs will be identified as the DAEC post defueled technical specifications (PDTS). Following the June 20, 2019, submittal, the licensee supplemented the original application by letters dated September 12, 2019, and November 4, 2019. NEDA performed an analysis of a fuel handling accident (FHA) in the spent fuel pool (SFP). This analysis determined that, following a decay period of 19 days, control building emergency ventilation is not required to maintain FHA dose consequences for control room occupants below the acceptance criteria of 10 CFR 50.67(b)(2)(iii). Consequently, NEDA hereby requests supplemental changes to the DAEC TSs to reflect the revised FHA analysis. Specifically, those TSs associated with control building emergency ventilation are proposed for deletion by this supplemental submittal.

The proposed supplemental changes to the DAEC TSs are in accordance with 10 CFR 50.36(c)(1) through (c)(5). The proposed supplemental changes also include administrative changes to content format and revised page numbering. The TS Table of Contents will be revised accordingly.

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

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

Response: No.

The proposed changes would not take effect until DAEC has certified to the NRC that it has permanently ceased operation and entered a permanently defueled condition. Because the 10 CFR part 50 license for DAEC will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR part 50.82(a)(1) submitted, as specified in 10 CFR part 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. DAEC's accident analyses are contained in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). In a permanently defueled condition, the only credible UFSAR described accident that remains is the Fuel Handling Accident (FHA). Other Chapter 15 accidents will no longer be applicable to a permanently defueled reactor.

The UFSAR-described FHA analyses for DAEC shows that, following the required decay time after reactor shutdown and provided the SFP water level requirement of TS LCO [limiting condition for operation] 3.7.8 is met, the dose consequences are acceptable without relying on secondary containment or the Standby Gas Treatment System. The control building envelop is credited for reduction of operator dose. Consequently, the TS requirements for the Standby Filter Unit and Control Building Chillers are retained.

The probability of occurrence of previously evaluated accidents is not increased, since safe storage and handling of fuel will be the only operations performed, and therefore, bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in the permanently defueled condition. This significantly reduces the scope of applicable accidents. The deletion of TS definitions and rules of usage and application requirements that will not be applicable in a defueled condition has no impact on facility SSCs [structures, system, and components] or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shut down and defueled DAEC has no impact on the remaining applicable DBA [design-basis accident].

The removal of LCOs or SRs [surveillance requirements] that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the permanently defueled condition.

The proposed changes, as supplemented, would not take effect until DAEC has certified to the NRC that it has permanently ceased operation, entered a permanently defueled condition, and a period of 19 days has transpired since shutdown. Because the 10 CFR part 50 license for DAEC will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR part 50.82(a)(1) submitted, as specified in 10 CFR part 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. DAEC's accident analyses are contained in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). In a permanently defueled condition, the only credible UFSAR described accident that remains is the Fuel Handling Accident (FHA). Other Chapter 15 accidents will no longer be applicable to a permanently defueled reactor.

The UFSAR-described FHA analyses for DAEC shows that, provided the SFP water level requirement of TS LCO 3.7.8 is met, the dose consequences are acceptable without relying on secondary containment or the Standby Gas Treatment System.

Once the DAEC has permanently shut down and defueled, the only credible FHA is a fuel drop in the SFP. NEDA performed an analysis of the SFP FHA. This analysis determined that, following a decay period of 19 days, Control Building emergency ventilation is not required to maintain FHA dose consequences for control room occupants below the acceptance criteria of 10 CFR 50.67(b)(2)(iii). Consequently, the TS requirements for the systems supporting the Control Building emergency ventilation are proposed for deletion.

The probability of occurrence of previously evaluated accidents is not increased, since safe storage and handling of fuel will be the only operations performed, and therefore, bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in the permanently defueled condition. This significantly reduces the scope of applicable accidents. The deletion of TS definitions and rules of usage and application requirements that will not be applicable in a defueled condition has no impact on facility SSCs or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shut down and defueled DAEC has no impact on the remaining applicable DBA.

The removal of LCOs or SRs that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the permanently defueled condition.

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

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

   Response: No.

   The proposed changes to delete or modify certain DAEC Operating License, TS, and current licensing bases (CLB) have no impact on facility SSCs affecting the safe storage of spent irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of the spent irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor, or only to the prevention, diagnosis, or mitigation of reactor related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shut down and defueled.

   The proposed modification or deletion of requirements of the DAEC Operating License, TS, and CLB do not affect systems credited in the accident analysis for the remaining credible DBA at DAEC. The proposed Operating License and PDTS will continue to require proper control and monitoring of safety significant parameters and activities. The TS regarding SFP water level and spent fuel storage is retained to preserve the current requirements for safe storage of irradiated fuel. The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding, spent fuel racks, SFP integrity, and SFP water level). Since extended operation in a defueled condition and safe fuel handling will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

   The proposed changes, as supplemented, to delete or modify certain DAEC TS, and current licensing bases (CLB) have no impact on facility SSCs affecting the safe storage of spent irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of the spent irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor, or only to the prevention, diagnosis, or mitigation of reactor related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shut down and defueled.

The proposed modification or deletion of requirements of the DAEC TS, and CLB do not affect systems credited in the accident analysis for the remaining credible DBA at DAEC. The proposed TS will continue to require proper control and monitoring of safety significant parameters and activities. The TS regarding SFP water level is retained to preserve the current requirements for safe storage of irradiated fuel. The proposed amendment, as supplemented, does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding, spent fuel racks, SFP integrity, and SFP water level). Since extended operation in a defueled condition and safe fuel handling will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

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

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

   Response: No.

   The proposed changes are to delete or modify certain Operating License, TS and CLB once the DAEC facility has been permanently shut down and defueled. As specified in 10 CFR 50.82(a)(2), the 10 CFR 50 license for DAEC will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel following submittal of the certifications required by 10 CFR 50.82(a)(1). As a result, the occurrence of certain design basis postulated accidents are no longer considered credible when the reactor is permanently defueled.

   The only remaining credible UFSAR described accident is a[n] FHA. The proposed changes do not adversely affect the inputs or assumptions of any of the design basis analyses that impact the FHA.

   The proposed changes are limited to those portions of the Operating License, TS, and CLB that are not related to the safe storage of irradiated fuel. The requirements proposed to be revised or deleted from the Operating License, TS, and CLB are not credited in the existing accident analysis for the remaining postulated accident (i.e., FHA); and, as such, do not contribute to

> the margin of safety associated with the accident analysis. Certain postulated DBAs involving the reactor are no longer possible because the reactor will be permanently shut down and defueled and DAEC will no longer be authorized to operate the reactor.
>
> The proposed changes, as supplemented, are to delete or modify certain TS and CLB once the DAEC facility has been permanently shut down and defueled and a period of no less than 19 days has transpired since shutdown. As specified in 10 CFR 50.82(a)(2), the 10 CFR 50 license for DAEC will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel following submittal of the certifications required by 10 CFR 50.82(a)(1). As a result, the occurrence of certain design basis postulated accidents are no longer considered credible when the reactor is permanently defueled.
>
> The only remaining credible UFSAR described accident is a[n] FHA. Further, an FHA in the reactor core is no longer credible. An FHA in the SFP is the only remaining credible accident. NEDA has performed a revised analysis for an FHA in the SFP. This analysis determined that, following a decay period of 19 days, Control Building emergency ventilation is not required to maintain FHA dose consequences for control room occupants below the acceptance criteria of 10 CFR 50.67(b)(2)(iii). Consequently, TS LCOs and SRs associated with CBEV [Control Building emergency ventilation] and support equipment are proposed for deletion. The proposed changes, as supplemented, do not adversely affect the inputs or assumptions of the revised FHA analysis.
>
> The proposed changes, as supplemented, are limited to those portions of the TS, and CLB that are not related to the safe storage of irradiated fuel. The requirements proposed to be revised or deleted from the TS, and CLB are not credited in the existing accident analysis for the remaining postulated accident (i.e., FHA in the SFP); and, as such, do not contribute to the margin of safety associated with the accident analysis. Certain postulated DBAs involving the reactor are no longer possible because the reactor will be permanently shut down and defueled and DAEC will no longer be authorized to operate the reactor.
>
> Therefore, the proposed changes, as supplemented, have 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: Steven Hamrick, Managing Attorney - Nuclear, Florida Power Light Company, P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: September 25, 2019, as supplemented by letter dated November 4, 2019. Publicly-available versions are in ADAMS under Accession Nos. ML19290G447, and ML19308A085, respectively.

Description of amendment request: The amendment would delete the DAEC Operating License Condition 2.C.(3), "Fire Protection Program," which requires that NEDA implement and maintain a fire protection program that complies with the requirements of 10 CFR 50.48(a) and 10 CFR 50.48(c). NEDA will maintain a Fire Protection Program in accordance with 10 CFR 50.48(f), as required for licensees that have submitted certification of permanent cessation of operations.

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 alter, degrade or prevent action described or assumed in any accident in the UFSAR [updated final safety analysis report] from being performed. The proposed change does not alter any assumptions previously made in evaluating radiological consequences. The proposed change does not affect the integrity of any fission product barrier.

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 alter any safety limits or safety analysis assumptions associated with the operation of the plant. The proposed change does not introduce any new accident initiators, nor does the change reduce or adversely affect the capabilities of any plant structure or system in the performance of its safety function.

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

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

   Response: No.

   The proposed change does not alter the manner in which safety limits or limiting safety system settings are determined. The safety analysis acceptance criteria are not affected by the proposed change. The proposed change does not change the design function of any equipment assumed to operate in the event of an accident.

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

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

<u>Attorney for licensee</u>: Steven Hamrick, Managing Attorney - Nuclear, Florida Power Light Company, P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Nancy L. Salgado.

Northern States Power Company - Minnesota (NSPM), Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant (PINGP), Unit Nos.1 and 2, Goodhue County, Minnesota

Date of amendment request: October 7, 2019. A publicly-available version is in ADAMS under Accession No. ML19280B335.

Description of amendment request: The amendments would revise technical specifications (TSs) for the PINGP, Units 1 and 2. The proposed change revises TS 5.5.14, "Containment Leakage Rate Testing Program," to increase the containment integrated leakage rate test program Type A test interval from 10 to 15 years and extend the containment isolation valve Type C leakage rate test frequency from 60 to up to 75 months.

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 adopts the NRC-accepted guidelines of NEI [Nuclear Energy Institute] 94-01 for the development of the NSPM performance-based containment testing program for PINGP Units 1 and 2. NEI 94-01 allows, based on risk and performance, an extension of the Type A and Type C containment leak test intervals. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses.

   The findings of the PINGP risk assessment confirm the general findings of previous studies that the risk impact with extending the containment leak rate is small. In accordance with the guidance provided in Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” an extension of the leak test interval in accordance with NEI 94-01, Revision 3-A results in an estimated change within the very small change region.

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

Therefore, the proposed 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 to implement a performance-based containment testing program, associated with integrated leakage rate test frequency, does not change the design or operation of structures, systems, or components of the plant. The proposed change would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses. There are no accident initiators created or affected by this change.

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.

Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. The proposed change to implement a performance-based containment testing program, associated with integrated leakage rate test and local leak rate

> testing frequency, does not affect plant operations, design functions, or any analysis that verifies the capability of a structure, system, or component of the plant to perform a design function. In addition, this change does not affect safety limits, limiting safety system setpoints, or limiting conditions for operation.
>
> The specific requirements and conditions of the TS Containment Leakage 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 the TSs is maintained. This ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met with the acceptance of this proposed change since these are not affected by implementation of a performance-based containment testing program.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: September 30, 2019. A publicly-available version is in ADAMS under Accession No. ML19273A953.

Description of amendment request: The amendment request proposes changes to the Combined License (COL) Numbers NPF-91 and NPF-92 for VEGP, Units 3 and 4, and proposes to depart from Updated Final Safety Analysis Report (UFSAR) Tier 2 information (which includes the plant-specific Design Control Document (DCD) Tier 2 information). The proposed changes involve related changes to plant-specific Tier 1 information, with corresponding changes to the associated COL Appendix C information, and involves related changes to COL Appendix A, Technical Specifications. Specifically, the requested amendment proposes changes to reflect revisions in the design parameters of (a) the maximum stroke times for the automatic depressurization system (ADS) Stages 1, 2 and 3 valves, (b) the minimum effective flow areas for the ADS Stages 2 and 3 valves, and (c) the core makeup tank minimum volume. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR part 52, appendix D, design certification rule is also requested for the plant-specific DCD Tier 1 material departures.

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

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

   Response: No.

   The proposed revisions to the automatic depressurization system (ADS) and core makeup tank (CMT) design parameters have been found to continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The ADS and CMT design parameters are not an initiator of any accident analyzed in the Updated Final Safety Analysis Report (UFSAR), nor do the changes involve an interface with any structure, system or component (SSC) accident initiator or initiating sequence of events, and thus, the probabilities of the accidents evaluated in the UFSAR are not affected. The proposed changes do not involve a change to any mitigation sequence or the predicted radiological releases due to postulated accident conditions, thus, the consequences of the accidents evaluated in the UFSAR are not affected.

The UFSAR describes the analyses of various design basis transients and accidents to demonstrate compliance of the design with the acceptance criteria for these events. The acceptance criteria for the various events are based on meeting the relevant regulations, general design criteria, and the Standard Review Plan, and are a function of the anticipated frequency of occurrence of the event and potential radiological consequences to the public. The revised accident analyses maintain their plant conditions, and thus their frequency designation and consequence level as previously evaluated.

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

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

   Response: No.

   The proposed revisions to the ADS and CMT design parameters have been found to continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The proposed revisions to the ADS and CMT design parameters do not change the function of the related systems, and thus, the changes do not introduce a new failure mode, malfunction or sequence of events that could adversely affect safety or safety-related equipment.

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

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

   Response: No.

   The proposed revisions to the ADS and CMT design parameters have been found to continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The proposed revisions to the ADS and CMT design parameters does not change the function of the related systems nor significantly affect the margins provided by the systems. No safety analysis or design basis

> acceptance limit/criterion is challenged or exceeded by the requested changes.
>
> 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: M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Victor Hall.

IV. Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the *Federal Register* on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: October 23, 2019. A publicly-available version is in ADAMS under Accession No. ML19296C538.

Description of amendment request: The amendments would revise the Watts Bar Nuclear Plant, Units 1 and 2, Technical Specification Table 3.3.5-1, “LOP [Loss of Power] DG [Diesel Generator] Start Instrumentation,” Function 5, “6.9 kV [kilovolt] Emergency Bus Undervoltage (Unbalanced Voltage),” to correct the values for the allowable value for the unbalanced voltage relay (UVR) low trip voltage, the allowable value for the UVR high trip time delay, and the trip setpoint for the UVR high trip time delay.

Date of publication of individual notice in *Federal Register*: November 6, 2019 (84 FR 59846).

Expiration date of individual notice: December 6, 2019 (public comments); January 6, 2020 (hearing requests).

V. 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.

<u>DTE Electric Company, Docket No. 50-341, Fermi 2, Monroe County, Michigan</u>

<u>Date of amendment request</u>: February 8, 2019

<u>Brief description of amendment</u>: The amendment adopted Technical Specifications Task Force (TSTF)-564, "Safety Limit MCPR (Minimum Critical Power Ratio)," Revision 2, and revises the Fermi 2 technical safety limit on MCPR to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for a safety limit. In addition, TS 5.6.5, Core Operating Limits Report (COLR), was revised to require the current safety limit MCPR value to be included in the cycle specific COLR.

<u>Date of issuance</u>: November 5, 2019.

<u>Effective date</u>: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 214. A publicly-available version is in ADAMS under Accession No. ML19189A004; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF-43: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: April 9, 2019 (84 FR 14144).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 2019.

No significant hazards consideration comments received: No.

NextEra Energy Seabrook, LLC, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: October 4, 2018, as supplemented by letter dated September 30, 2019.

Description of amendment request: The amendment revised the technical specifications to adopt changes provided in Technical Specifications Task Force (TSTF)-234, "Add Action for More than One (Digital Rod Position Indication) [D]RPI Inoperable"; TSTF-547, "Clarification of Rod Position Requirements"; and made various other changes to align the Seabrook TSs more closely with NUREG-1431, "Standard Technical Specifications Westinghouse Plants."

Date of issuance: November 18, 2019.

Effective date: As of its date of issuance and shall be implemented by May 28, 2020.

Amendment No.: 162. A publicly-available version is in ADAMS under Accession No. ML19224A563; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF-86: The amendment revised the Renewed Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: April 9, 2019 (84 FR 14151). The supplemental letter dated September 30, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 2019.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 20, 2018, as supplemented by letters dated April 29, 2019 and August 5, 2019.

Brief description of amendment: The amendments added a condition to the PINGP, Units 1 and 2, renewed facility operating licenses to allow the implementation of 10 CFR 50.69, "Risk informed categorization and treatment of structures, systems and components for nuclear power reactors."

Date of issuance: November 12, 2019.

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

Amendment Nos.: 230 (Unit 1); 218 (Unit 2). A publicly-available version is in ADAMS under Accession No. ML19276F684; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

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

Date of initial notice in *Federal Register*: September 11, 2018 (83 FR 45986). The supplemental letters dated April 29, 2019 and August 5, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 2019.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 2, 2018, as supplemented by letter dated December 4, 2018.

Brief description of amendment: The amendments revised the design basis accident dose threshold for designation of certain fuel handling equipment as Quality Type I (safety-related) to greater than 10 percent of the dose limits specified in 10 CFR part 100, "Reactor Site Criteria."

Date of issuance: November 7, 2019.

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

Amendment Nos.: 229 (Unit 1); 217 (Unit 2). A publicly-available version is in ADAMS under Accession No. ML19232A151; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-42 and DPR-60: The amendments revised the Updated Safety Analysis Report.

Date of initial notice in *Federal Register*: January 31, 2019 (84 FR 812).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 2019.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 27, 2019.

Brief description of amendment: The amendment adopted Technical Specifications Task Force (TSTF) Traveler TSTF-546, "Revise APRM [Average Power Range Monitor] Channel Adjustment Surveillance Requirement," which revises the Hope Creek Generating Station technical specification surveillance requirement to verify that calculated power is no more than 2 percent greater than the APRM channel output. This change revised the surveillance requirement to distinguish between APRM indications that are consistent with the accident analyses and those that provide additional margin.

Date of issuance: November 7, 2019.

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

Amendment No.: 220. A publicly-available version is in ADAMS under Accession No. ML19289A886; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

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

Date of initial notice in *Federal Register*: April 9, 2019 (84 FR 14152).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 2019.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC and Exelon Generation Company, LLC, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: February 4, 2019, as supplemented by letter dated June 11, 2019.

Brief description of amendments: The amendments revised the Technical Specification requirements on control and shutdown rods and rod and bank position indication, consistent with NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-547, Revision 1, "Clarification of Rod Position Requirements," dated March 4, 2016.

Date of issuance: November 18, 2019.

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

Amendment Nos.: 330 (Unit No. 1) and 311 (Unit No. 2). A publicly-available version is in ADAMS under Accession No. ML19275D694; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-70 and DPR-75: The amendments revised the Renewed Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: March 26, 2019 (84 FR 11339). The supplemental letter dated June 11, 2019, 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 November 18, 2019.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Appling County, Georgia

Date of amendment request: July 23, 2019.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) actions for inoperable residual heat removal (RHR) shutdown cooling subsystems in the RHR shutdown cooling system limiting conditions for operation. The proposed changes are based on Technical Specifications Task Force (TSTF) traveler TSTF-566, Revision 0, "Revise Actions for Inoperable RHR Shutdown Cooling Subsystems," dated January 19, 2018.

Date of issuance: November 13, 2019.

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

Amendment Nos.: 300 (Unit No. 1) and 245 (Unit No. 2). A publicly-available version is in ADAMS under Accession No. ML19267A023; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-57 and NPF-5: The amendments revised the Renewed Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: September 10, 2019 (84 FR 47551).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 2019.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: February 1, 2019.

Brief description of amendments: The amendments adopted Technical Specifications Task Force (TSTF) Traveler TSTF-563, Revision 0, "Revise Instrument Testing Definitions to Incorporate the Surveillance Frequency Control Program."

Date of issuance: November 18, 2019.

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

Amendment Nos.: 347 (Unit 1) and 341 (Unit 2). A publicly-available version is in ADAMS under Accession No. ML19281B554; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Renewed Facility Operating License Nos. DPR-77 and DPR-79: The amendments revised the Renewed Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: April 9, 2019 (84 FR 14153).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 2019.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: November 19, 2018, as supplemented by letter dated August 22, 2019.

Brief description of amendments: The amendments approved installation of two non-safety-related water headers within a safety-related flood protection dike.

Date of issuance: November 13, 2019.

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

Amendment Nos.: 283 (Unit No. 1) and 266 (Unit No. 2). A publicly-available version is in ADAMS under Accession No. ML19274C998; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License No. NPF-4 and NPF-7: The amendments revised the Renewed Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: March 26, 2019 (84 FR 11342). The supplement dated August 22, 2019, 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 November 13, 2019.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 25th day of November 2019.

For the Nuclear Regulatory Commission.

Craig G. Erlanger, Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.

[FR Doc. 2019-25972 Filed: 12/2/2019 8:45 am; Publication Date: 12/3/2019]