entities, persons, products, and offerings.

7. United States Postal Service, Office of the Regional Director, Atlanta, Georgia (DAA–0028–2018–0001, 6 items, 6 temporary items). Dedication files and site selection files of individual post offices in Florida, Georgia, North Carolina, South Carolina, and Puerto Rico. Includes personnel records and routine organization data.

Laurence Brewer,

Chief Records Officer for the U.S. Government.

[FR Doc. 2018–11987 Filed 6–4–18; 8:45 am]

BILLING CODE 7515–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2018–0105]

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from May 8, 2018, to May 21, 2018. The last biweekly notice was published on May 22, 2018.

DATES: Comments must be filed by July 5, 2018. A request for a hearing must be filed by August 6, 2018.

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

- *Federal Rulemaking* website: Go to *http://www.regulations.gov* and search for Docket ID NRC–2018–0105. Address questions about NRC dockets to Jennifer Borges; telephone: 301–287–9127; email: *Jennifer.Borges@nrc.gov.* For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.
- *Mail comments to:* May Ma, Office of Administration, Mail Stop: TWFN–7–A60M, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

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

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

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC–2018–0105, 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 website:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2018–0105.
- *NRC's Agencywide Documents Access and Management System (ADAMS):* You may obtain publicly-available documents online in the ADAMS Public Documents collection at *http://www.nrc.gov/reading-rm/adams.html.* To begin the search, select ''ADAMS Public Documents'' and then select ''Begin Web-based ADAMS Search.'' For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr.resource@nrc.gov.* The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document.
- *NRC's PDR:* You may examine and purchase copies of public documents at the NRC's PDR, Room O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID NRC–2018–0105, facility name, unit number(s), plant docket number, application date, and subject in your comment submission.

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

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

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

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

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

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

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

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

As required by 10 CFR 2.309(d) the petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements for standing: (1) The name, address, and telephone number of the petitioner; (2) the nature of the petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the petitioner's interest.

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

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

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

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

A State, local governmental body, Federally-recognized Indian Tribe, or agency thereof, may submit a petition to the Commission to participate as a party under 10 CFR 2.309(h)(1). The petition should state the nature and extent of the petitioner's interest in the proceeding. The petition should be submitted to the Commission no later than 60 days from the date of publication of this notice August 6, 2018. 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 website at *http://www.nrc.gov/site-help/e-submittals.html.* Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by email 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 website at *http://www.nrc.gov/site-help/e-submittals/getting-started.html.* Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit adjudicatory documents. Submissions must be in Portable Document Format (PDF). Additional guidance on PDF submissions is available on the NRC's public website at *http://www.nrc.gov/site-help/electronic-sub-ref-mat.html.* A filing is considered complete at the time the document is submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an email notice confirming receipt of the document. The E-Filing system also distributes an email notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the 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 website at *http://www.nrc.gov/site-help/e-submittals.html,* by email to *MSHD.Resource@nrc.gov,* or by a toll-free call at 1–866–672–7640. The NRC 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.

Duke Energy Carolinas, LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2 (MNS), Mecklenburg County, North Carolina

Date of amendment request: December 8, 2017. A publicly-available version is in ADAMS under Accession No. ML17352A404.

Description of amendment request: The amendments would modify the MNS, Unit Nos. 1 and 2 Updated Final Safety Analysis Report (UFSAR) to describe the methodology and results of the analyses performed to evaluate the protection of the plant's structures, systems, and components from tornado-generated missiles.

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

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

Response: No.

The proposed changes to the MNS UFSAR constitutes a license amendment to incorporate use of a Nuclear Regulatory Commission (NRC) approved probabilistic methodology to assess the need for additional positive (physical) tornado missile protection of specific features at the MNS site. The UFSAR changes will reflect use of the Electric Power Research Institute (EPRI) Topical Report ''Tornado Missile Risk

Evaluation Methodology'' (EPRI NP–2005), Volumes I and II. As noted in the NRC Safety Evaluation Report on this topic dated October 26, 1983, the current licensing criteria governing tornado missile protection are contained in NUREG–0800, Sections 3.5.1.4 and 3.5.2. These criteria generally specify that safety-related systems, structures and components be provided positive tornado missile protection (barriers) from the maximum credible tornado threat. However, NUREG–0800 includes acceptance criteria permitting relaxation of the above deterministic guidance, if it can be demonstrated that the probability of damage to unprotected essential safety-related features is sufficiently small.

As permitted in NUREG–0800 sections, the combined probability will be maintained below an allowable level, *i.e.,* an acceptance criterion threshold, which reflects an extremely low probability of occurrence. The approach assumes that if the sum of the individual probabilities calculated for tornado missiles striking and damaging portions of important systems, structures or components is greater than or equal to 1×10^{-6} per year per unit, then installation of unique missile barriers would be needed to lower the total cumulative probability below the acceptance criterion of 1×10^{-6} per year per unit.

With respect to the probability of occurrence or the consequences of an accident previously evaluated in the UFSAR, the possibility of a tornado reaching the site and causing damage to plant structures, systems and components is considered in the MNS UFSAR.

The change being proposed does not affect the probability that the natural phenomenon (a tornado) will reach the plant, but from a licensing basis perspective, the change does affect the probability that missiles generated by the winds of the tornado might strike and damage certain plant structures, systems and components. There are a limited number of safety-related components that could theoretically be struck and damaged by tornadogenerated missiles. The probability of tornado-generated missile strikes on important to safety structures, systems and components is what was analyzed using the probabilistic methods discussed above. The combined probability of damage will be maintained below an extremely low acceptance criterion to ensure overall plant safety. The proposed change is not considered to constitute a significant increase in the probability of occurrence or the consequences of an accident, due to the extremely low probability of damage due to tornado-generated missiles and thus an extremely low probability of a radiological release.

The results of the analysis documented in this [license amendment request (LAR)] are below the acceptance criterion of 1×10^{-6} per year per unit. 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 changes to the MNS UFSAR incorporate use of a NRC approved probabilistic methodology to assess the need for additional positive (physical) tornado missile protection for specific features. This will not change the design function or operation of any structure, system or component. This proposed change does not involve any plant modifications. There are no new credible failure mechanisms, malfunctions or accident initiators not considered in the design and licensing bases for MNS. The proposed change involves an already established tornado design basis event and the tornado event is explicitly considered in the MNS UFSAR.

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

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

Response: No.

The existing licensing basis for MNS for protecting safety-related, safe shutdown equipment from tornado generated missiles is to provide positive missile barriers for all safety-related structures, systems and components. The proposed change recognizes that there is an extremely low probability, below an established acceptance limit, that a limited subset of the safety-related, safe shutdown structures, systems and components could be struck and consequently damaged. The change from requiring protection of all safety-related, safety shutdown structures, systems and components from tornadogenerated missiles, to only a subset of equipment, is not considered to constitute a significant decrease in the margin of safety due to that extremely low probability of occurrence of tornado-generated missile strikes and consequential damage.

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

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

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

NRC Branch Chief: Michael T. Markley.

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: April 5, 2018. A publicly-available version is in ADAMS under Accession No. ML18099A130.

Description of amendment request: The proposed amendment would revise the licensing basis, by the addition of a license condition, to allow for the implementation of the provisions of 10 CFR 50.69, ''Risk-informed categorization and treatment of structures, systems, and components [SSCs] for nuclear power reactors.''

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

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

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

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 will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

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

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

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

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

NRC Acting Branch Chief: Brian W. Tindell.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: March 12, 2018, as supplemented by letter dated April 26, 2018. Publicly-available versions are in ADAMS under Accession Nos. ML18071A319 and ML18117A493, respectively.

Description of amendment request: The amendment would revise the Arkansas Nuclear One, Unit No. 1 Technical Specifications (TSs) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)-425, Revision 3, ''Relocate Surveillance Frequencies to Licensee Control—RITSTF [Risk-informed TSTF] Initiative 5b.'' Additionally, the change would add a new program, the Surveillance Frequency Control Program, to TS Section 5.5, ''Programs and Manuals.''

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

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

Response: No.

The proposed change relocates the specified frequencies for periodic surveillance requirements (SRs) to licensee control under a new Surveillance Frequency Control Program [SFCP]. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications (TSs) for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the SRs, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

Response: No.

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (*i.e.,* no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed 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 the margin of safety?

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Entergy will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI [Nuclear Energy Institute] 04–10, Rev. 1 in accordance with the TS SFCP. NEI 04–10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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: Anna Vinson Jones, Senior Counsel, Entergy Services, Inc., 101 Constitution Avenue NW, Suite 200 East, L–ENT–WDC, Washington, DC 20001.

NRC Branch Chief: Robert J. Pascarelli.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: February 6, 2018, as supplemented by letter dated March 26, 2018. Publicly-available versions are in ADAMS under Accession Nos. ML18038B354, and ML18085A816, respectively.

Description of amendment request: The amendment would revise the Arkansas Nuclear One, Unit No. 2 Technical Specifications (TSs) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specifications Task Force (TSTF)-425, Revision 3, ''Relocate Surveillance Frequencies to Licensee Control—RITSTF [Risk-Informed TSTF] Initiative 5b.'' The amendment would also add a new program, the Surveillance Frequency Control Program, to TS Section 6.0, ''Administrative Controls.''

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

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

Response: No.

The proposed change relocates the specified frequencies for periodic Surveillance Requirements (SRs) to licensee control under a new Surveillance Frequency Control Program (SFCP). Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the TSs for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the SRs, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

Response: No.

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (*i.e.,* no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed 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 the margin of safety?

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Entergy will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI [Nuclear Energy Institute] 04–10, Rev. 1, in accordance with the TS SFCP. NEI 04–10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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: Anna Vinson Jones, Senior Counsel, Entergy Services, Inc., 101 Constitution Avenue NW, Suite 200 East, L–ENT–WDC, Washington, DC 20001.

NRC Branch Chief: Robert J. Pascarelli.

Entergy Operations, Inc.; System Energy Resources, Inc.; Cooperative Energy, A Mississippi Electric Cooperative; and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit No. 1, Claiborne County, Mississippi

Date of amendment request: April 10, 2018. A publicly-available version is in ADAMS under Accession No. ML18100B304.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to adopt Technical Specifications Task Force (TSTF) Traveler TSTF–542, Revision 2, ''Reactor Pressure Vessel Water Inventory Control.'' The proposed change would replace existing TS requirements related to ''operations with a potential for draining the reactor vessel'' (OPDRVs) with new requirements on reactor pressure vessel (RPV) water inventory control (WIC) to protect Safety Limit 2.1.1.3. Safety Limit 2.1.1.3 requires reactor vessel water level to be greater than the top of active irradiated fuel.

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 replaces existing TS requirements related to OPDRVs with new requirements on RPV WIC that will protect Safety Limit 2.1.1.3. Draining of RPV water inventory in Mode 4 (*i.e.,* cold shutdown) and Mode 5 (*i.e.,* refueling) is not an accident previously evaluated and, therefore, replacing the existing TS controls to prevent or mitigate such an event with a new set of controls has no effect on any accident previously evaluated. RPV water inventory control in Mode 4 or Mode 5 is not an initiator of any accident previously evaluated. The existing OPDRV controls or the proposed RPV WIC controls are not mitigating actions assumed in any accident previously evaluated.

The proposed change reduces the probability of an unexpected draining event (which is not a previously evaluated accident) by imposing new requirements on the limiting time in which an unexpected draining event could result in the reactor vessel water level dropping to the top of the active fuel (TAF). These controls require cognizance of the plant configuration and control of configurations with unacceptably short drain times. These requirements reduce the probability of an unexpected draining event. The current TS requirements are only mitigating actions and impose no requirements that reduce the probability of an unexpected draining event.

The proposed change reduces the consequences of an unexpected draining event (which is not a previously evaluated accident) by requiring an Emergency Core Cooling System (ECCS) subsystem to be operable at all times in Modes 4 and 5. The current TS requirements do not require any water injection systems, ECCS or otherwise, to be Operable in certain conditions in Mode 5. The change in requirement from two ECCS subsystems to one ECCS subsystem in Modes 4 and 5 does not significantly affect the consequences of an unexpected draining event because the proposed Actions ensure equipment is available within the limiting drain time that is as capable of mitigating the event as the current requirements. The proposed controls provide escalating compensatory measures to be established as calculated drain times decrease, such as verification of a second method of water injection and additional confirmations that containment and/or filtration would be available if needed.

The proposed change reduces or eliminates some requirements that were determined to be unnecessary to manage the consequences of an unexpected draining event, such as automatic initiation of an ECCS subsystem and control room ventilation. These changes do not affect the consequences of any accident previously evaluated since a draining event in Modes 4 and 5 is not a previously evaluated accident and the requirements are not needed to adequately respond to a draining event.

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

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

Response: No.

The proposed change replaces existing TS requirements related to OPDRVs with new requirements on RPV WIC that will protect Safety Limit 2.1.1.3. The proposed change will not alter the design function of the equipment involved. Under the proposed change, some systems that are currently required to be operable during OPDRVs would be required to be available within the limiting drain time or to be in service depending on the limiting drain time. Should those systems be unable to be placed into service, the consequences are no different than if those systems were unable to perform their function under the current TS requirements.

The event of concern under the current requirements and the proposed change is an unexpected draining event. The proposed change does not create new failure mechanisms, malfunctions, or accident initiators that would cause a draining event or a new or different kind of accident not previously evaluated or included in the design and licensing bases.

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 replaces existing TS requirements related to OPDRVs with new requirements on RPV WIC. The current requirements do not have a stated safety basis and no margin of safety is established in the licensing basis. The safety basis for the new requirements is to protect Safety Limit 2.1.1.3. New requirements are added to determine the limiting time in which the RPV water inventory could drain to the top of the fuel in the reactor vessel should an unexpected draining event occur. Plant configurations that could result in lowering the RPV water level to the TAF within one hour are now prohibited. New escalating compensatory measures based on the limiting drain time replace the current controls. The proposed TS establish a safety margin by providing defense-in-depth to ensure that the Safety Limit is protected and to protect the public health and safety. While some less restrictive requirements are proposed for plant configurations with long calculated drain times, the overall effect of the change is to improve plant safety and to add safety margin.

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

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

Attorney for licensee: Anna Vinson Jones, Senior Counsel/Legal Department, Entergy Services, Inc., 101 Constitution Avenue NW, Suite 200 East, Washington, DC 20001.

NRC Branch Chief: Robert J. Pascarelli.

Entergy Operations, Inc.; System Energy Resources, Inc.; Cooperative Energy, A Mississippi Electric Cooperative; and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit No. 1 (GGNS), Claiborne County, Mississippi

Date of amendment request: April 27, 2018. A publicly-available version is in ADAMS under Accession No. ML18117A514.

Description of amendment request: The proposed amendment would revise the Emergency Plan to adopt the Nuclear Energy Institute's (NEI's) revised Emergency Action Level (EAL) scheme described in NEI 99–01, Revision 6, ''Development of Emergency Action Levels for Non-Passive Reactors'' (ADAMS Accession No. ML110240324), which has been endorsed by the NRC (ADAMS Accession No. ML12346A463).

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

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

Response: No.

The proposed changes to the GGNS EALs do not involve any physical changes to plant equipment or systems and do not alter the assumptions of any accident analyses. The proposed changes do not adversely affect accident initiators or precursors and do not alter design assumptions, plant configuration, or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems or components (SSCs) to perform intended safety functions in mitigating the consequences of an initiating event within the assumed acceptance limits.

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

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

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The changes do not challenge the integrity or performance of any safety-related systems. No plant equipment is installed or removed, and the changes do not alter the design, physical configuration, or method of operation of any plant SSC. Because EALs are not accident initiators and no physical changes are made to the plant, no new causal mechanisms are introduced.

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

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

Response: No.

Margin of safety is associated with the ability of the fission product barriers (*i.e.,* fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed changes do not impact operation of the plant and no accident analyses are affected by the proposed changes. The changes do not affect the Technical Specifications or the method of operating the plant. Additionally, the proposed changes will not relax any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis. The proposed changes do not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

Therefore, the 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: Anna Vinson Jones, Senior Counsel/Legal Department, Entergy Services, Inc., 101 Constitution Avenue NW, Suite 200 East, Washington, DC 20001.

NRC Branch Chief: Robert J. Pascarelli.

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois, and Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Date of amendment request: April 2, 2018. A publicly-available version is in ADAMS under Accession No. ML18092B081.

Description of amendment request: The proposed amendments would revise Technical Specification 3.2.3 to require that the axial flux difference be maintained within the limits specified in the core operating limits report during MODE 1 with reactor thermal power greater or equal to 50 percent. An associated change would also be made to the NOTE modifying surveillance 3.2.3.1.

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

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

Response: No.

The proposed amendment requires that the AFD [axial flux difference] be maintained within the limits specified in the COLR [core operating limits report] at-all-times during MODE 1 when reactor power is ≥50% RTP [reactor thermal power]. This requirement will ensure that all FRD [fuel rod design] performance criteria remain satisfied during ANS [American Nuclear Society] Condition II events (*i.e.,* Faults of Moderate Frequency); thus, ensuring the integrity of the fuel rod cladding. It is noted that maintaining AFD within the COLR limits at-all-times when ≥50% RTP is the normal operating practice as specified in plant procedures.

The proposed change will have no impact on accident initiators or precursors; does not alter accident analysis assumptions; does not involve any physical plant modifications that would alter the design or configuration of the facility, or the manner in which the plant is maintained; and does not impact the probability of operator error.

The proposed amendment will not impact the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an accident. All accident analysis acceptance criteria will continue to be met as the proposed change will not affect the source term, containment isolation function, or radiological release assumptions for any accident previously evaluated.

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

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

Response: No.

The proposed change formalizes the existing operating practice of maintaining the AFD within the limits specified in the COLR at-all-times during MODE 1 when reactor power is ≥ 50% RTP. This change ensures that all FRD performance criteria remain satisfied during ANS Condition II events. The ANS Condition II events have all been previously evaluated in the Updated Final Safety Analysis Report.

The proposed change does not involve a design change or other changes that would impact safety-related SSCs from performing their specified safety functions.

The proposed change does not result in the creation of any new accident precursors; does not result in changes to any existing accident scenarios; and does not introduce any operational changes or mechanisms that would create the possibility of a new or different kind of accident.

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

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

Response: No.

The proposed change to maintain the AFD within the limits specified in the COLR at-all-times during MODE 1 when reactor power is ≥ 50% RTP ensures that all FRD performance criteria remain satisfied during ANS Condition II events; and thus, will maintain the existing margin of safety related to FRD performance criteria and ensure the integrity of the fuel rod cladding. The AFD limits specified in the COLR have been established in accordance with the analysis approach described in NRC-approved Westinghouse Topical Reports.

In addition, this change will have no impact on the margin of safety associated with other reactor core safety parameters such as fuel hot channel factors, core power tilt ratios, loss of coolant accident peak cladding temperature and peak local power density.

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 requested amendments involve no significant hazards consideration.

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

NRC Branch Chief: David J. Wrona.

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

Date of amendment request: March 28, 2018. A publicly-available version is in ADAMS under Accession No. ML18087A293.

Description of amendment request: The amendment would revise Technical Specification (TS) 5.5.5.2.d, ''Provisions for SG [Steam Generator] Tube Inspection,'' and TS 5.5.5.2.f, ''Provisions for SG Tube Repair Methods.'' More specifically, TSs 5.5.5.2.d.5 and 5.5.5.2.f.3 would be simplified and clarified, respectively, without changing the intent of the specifications. Specification 5.5.5.2.f.3 would also be amended by changing the number of fuel cycles that Westinghouse Electric Company, LLC leak-limiting Alloy 800 sleeves may remain in operation.

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.

Proposed amendment of Technical Specification 5.5.5.2.d.5 to simplify the description of the required inspection region, and Technical Specification 5.5.5.2.f.3 to clarify that this specification is only applicable to sleeves installed in the steam generator tubesheet and change the number of fuel cycles that an Alloy 800 steam generator tubesheet sleeve may remain in service from five to eight fuel cycles of operation, does not affect structures, systems or components of the plant, plant operations, design functions or analyses that verify the capability of structures, systems or components to perform a design function. The proposed amendment does not increase the likelihood of steam generator tube sleeve leakage.

The proposed amendment of Technical Specification 5.5.5.2.d.5 to simplify the description of the required inspection region, makes it clear that the steam generator parent tube is to be inspected in the areas where the joints will be established prior to installation of the sleeve, regardless of the sleeve location. This proposed amendment does not change the intent of the specification.

The proposed amendment of TS 5.5.5.2.f.3 includes two changes. The first change would add the words ''installed in the hot-leg or cold-leg tubesheet region'' after the words ''An Alloy 800 sleeve'' to make it clear that the specification only applies to Alloy 800 tube sleeves installed in the steam generator tubesheet. The design of Alloy 800 sleeves installed in steam generator tube locations other than the tubesheet does not include a nickel band. For these sleeves, nondestructive examination methods have been demonstrated to be effective and limits on sleeve operating life are not necessary. This proposed amendment does not change the intent of the specification.

The second change to TS 5.5.5.2.f.3, increases the number of fuel cycles Alloy 800 tube sleeves installed in the tubesheet may remain in service. The leak-limiting Alloy 800 sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applied stresses and fatigue usage for the sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin of NRC Regulatory Guide 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes.''

The leak-limiting Alloy 800 sleeve depth-based structural limit is determined using NRC guidance and the pressure stress equation of ASME Code, Section III with margin added to account for the configuration of long axial cracks. Calculations show that a depth-based limit of 45 percent through-wall degradation is acceptable. However, Technical Specifications 5.5.5.2.c.2 and 5.5.5.2.c.3 provide additional margin by requiring an Alloy 800 sleeved tube to be plugged on detection of any flaw in the sleeve or in the pressure boundary portion of the original tube wall in the sleeve to tube joint. Degradation of the original tube adjacent to the nickel band of an Alloy 800 sleeve installed in the tubesheet, regardless of depth, would not prevent the sleeve from satisfying design requirements. Thus, flaw detection capabilities within the original tube adjacent to the sleeve nickel band are a defense-in-depth measure, and are not necessary in order to justify continued operation of the sleeved tube.

Evaluation of repaired steam generator tube testing and analysis indicates that there are no detrimental effects on the leak-limiting Alloy 800 sleeve or sleeved tube assembly from reactor coolant system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions that may be experienced at Beaver Valley Power Station, Unit No. 2. Westinghouse is not aware of, and has no knowledge of any reports of parent-tube stress corrosion cracking (SCC) in the sleeve roll joint region for any Westinghouse sleeve design.

The proposed increase in the number of fuel cycles Alloy 800 tube sleeves installed in the tubesheet may remain in service has no effect on sleeve operation or capability of the sleeve to perform its design function. The mechanical and leakage tests have confirmed that degradation of the parent tube adjacent to the nickel band will not prevent the sleeve from satisfying its design function.

Consequences of a hypothetical failure of the leak-limiting Alloy 800 sleeve and tube assembly are bounded by the current main steam line break and steam generator tube rupture accident analyses described in the Beaver Valley Power Station, Unit No. 2 Updated Final Safety Analysis Report. The total number of plugged steam generator tubes (including equivalency associated with installed sleeves) is required to be consistent with accident analysis assumptions. The sleeve and tube assembly leakage during plant operation is required to be within the allowable Technical Specification leakage limits and accident analysis assumptions.

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.

Proposed amendment of Technical Specification 5.5.5.2.d.5 to simplify the description of the required inspection region, and Technical Specification 5.5.5.2.f.3 to clarify that this specification is only applicable to sleeves installed in the steam generator tubesheet do not change the intent of these specifications, and do not affect the design function or operation of the tube sleeves. The proposed amendment of Technical Specification 5.5.5.2.f.3 to change the number of fuel cycles that an Alloy 800 steam generator tubesheet sleeve may remain in service from five to eight fuel cycles of operation, does not affect the design function or operation of the tube sleeves. Since these changes do not create any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design or licensing bases, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The leak-limiting Alloy 800 sleeves are designed using the applicable ASME Code, and therefore meet the objectives of the original steam generator tubing. As a result, the functions of the steam generator will not be significantly affected by the installation of the proposed sleeve. Therefore, the only credible failure modes for the sleeve and tube are to leak or rupture, which has already been evaluated. The continued integrity of the installed sleeve and tube assembly is periodically verified as required by the Technical Specifications, and a sleeved tube will be plugged on detection of a flaw in the sleeve or in the pressure boundary portion of the original tube wall in the sleeve to tube joint.

The proposed amendment to Technical Specification 5.5.5.2.f.3 increases the number of fuel cycles Alloy 800 tube sleeves installed in the tubesheet may remain in service to eight fuel cycles of operation. Implementation of this proposed amendment has no significant effect on either the configuration of the plant, the manner in which it is operated, or ability of the sleeve to perform its design function.

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

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

Response: No.

Proposed amendment of Technical Specification 5.5.5.2.d.5 to simplify the description of the required inspection region, and Technical Specification 5.5.5.2.f.3 to clarify that this specification is only applicable to sleeves installed in the steam generator tubesheet, do not change the intent of these requirements or reduce the margin of safety. The proposed amendment to Technical Specification 5.5.5.2.f.3 to change the number of fuel cycles that an Alloy 800 steam generator tubesheet sleeve may remain in service from five to eight fuel cycles of operation, does not affect a design basis or safety limit (that is, the controlling numerical value for a parameter established in the Updated Final Safety Analysis Report or the license) or reduce the margin of safety.

The proposed amendment to Technical Specification 5.5.5.2.f.3 increases the number of fuel cycles Alloy 800 tube sleeves installed in the tubesheet may remain in service to eight fuel cycles of operation. Implementation of this proposed amendment would not affect a design basis or safety limit or reduce the margin of safety. The repair of degraded steam generator tubes with leak-limiting Alloy 800 sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. Minimum reactor coolant system flow rate from the cumulative effect of repaired (sleeved) and plugged tubes will be greater than the flow rate limit established in the Technical Specification limiting condition for operation 3.4.1. The design safety factors utilized for the sleeves are consistent with the safety factors in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code used in the original steam generator design. Tubes with sleeves are subject to the same safety factors as the original tubes, which are described in the performance criteria for steam generator tube integrity in the existing Technical Specifications. The sleeve and portions of the installed sleeve and tube assembly that represent the reactor coolant pressure boundary will be monitored, and a sleeved tube will be plugged if a flaw is detected in the sleeve or in the pressure boundary portion of the original tube wall in the leak-limiting sleeve and tube assembly. Use of the previously-identified design criteria and design verification testing ensures that the margin of safety is not significantly different from the original steam generator tubes.

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

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

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

NRC Branch Chief: James Danna.

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

Date of amendment request: March 28, 2018. A publicly-available version is in ADAMS under Accession No. ML18087A095.

Description of amendment request: The amendment would revise Technical Specification (TS) 3/4.8.1, ''AC [Alternating Current] Sources—Operating''; specifically, ACTION b concerning one inoperable emergency diesel generator (EDG). The proposed change would remove the Salem Nuclear Generating Station, Unit No. 3 (Salem Unit 3), gas turbine generator and replace it with portable diesel generators.

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 removes the requirement for the Salem Unit 3 gas turbine generator (GTG) and replaces it with the supplemental power source during the existing extended allowable outage time for the A or B EDG. The emergency diesel generators are safety related components which provide backup electrical power supply to the onsite Safeguards Distribution System. The emergency diesel generators are not accident initiators; the EDGs are designed to mitigate the consequences of previously evaluated accidents including a loss of offsite power. (During normal operation, the proposed portable diesel generators will not be connected to the plant.)

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed change is consistent with safety analysis assumptions and resultant consequences.

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 removes the requirement for the Salem Unit 3 gas turbine generator (GTG) and replaces it with the supplemental power source during the existing extended allowable outage time for the A or B EDG. The proposed change does not alter or involve any design basis accident initiators. Equipment will be operated in the same configuration and manner that is currently allowed and designed for.

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

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

Response: No.

The proposed change does not alter the permanent plant design, including instrument set points, nor does it change the assumptions contained in the safety analyses. The proposed change does not impact the redundancy or availability requirements of offsite power supplies or change the ability of the plant to cope with station blackout [(SBO)] events.

The EDGs continue to meet their design requirements; there is no reduction in capability or change in design configuration. The EDG response to LOOP [loss of offsite power], LOCA [loss-of-coolant accident], SBO, or fire is not changed by this proposed amendment; there is no change to the EDG operating parameters. The remaining operable emergency diesel generators are adequate to supply electrical power to the onsite Safeguards Distribution System. The proposed change does not alter a design basis or safety limit; therefore it does not significantly reduce the margin of safety. The EDGs will continue to operate per the existing design and regulatory requirements.

Therefore, it is concluded that 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: Jeffrie J. Keenan, PSEG Nuclear LLC–N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: James G. Danna.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Unit Nos. 1 and 2 (SQN), Hamilton County, Tennessee

Date of amendment request: March 9, 2018, as supplemented by letter dated April 11, 2018. Publicly-available versions are in ADAMS under Accession Nos. ML18071A349 and ML18102B430, respectively.

Description of amendment request: The amendments would make changes to the SQN Essential Raw Cooling Water (ERCW) Motor Control Centers (MCCs) and revise the Updated Final Safety Analysis Report (UFSAR).

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

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

Response: No.

The proposed change does not alter the safety function of any structure, system, or component, does not modify the manner in which the plant is operated, and does not alter equipment out-of-service time. In addition, this request does not degrade the ability of the ERCW to perform its intended safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

Response: No.

The proposed change does not involve any physical changes to plant safety related structure, system or component or alter the modes of plant operation in a manner that is outside the bounds of the system design analyses. The proposed change to complete the design change for the removal of mechanical interlock device from the feeder breakers and tie breakers for the ERCW MCCs and to revise the ERCW System Description in Section 9.2.2.2 of the SQN UFSAR to describe the normal and alternate power sources for the ERCW system does not create the possibility for an accident or malfunction of a different type than any evaluated previously in SQN's UFSAR. The proposal does not alter the way any safety related structure, system or component functions and does not modify the manner in which the plant is operated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change to remove the mechanical interlock device from the feeder breakers and tie breakers for ERCW MCCs 1B–B and 2B–B and to revise the ERCW System Description in Section 9.2.2.2 of the SQN UFSAR to describe the normal and alternate power sources for the ERCW system does not reduce the margin of safety because ERCW will continue to perform its safety function. The design features provided by the mechanical interlock device are not described in the SQN UFSAR, are not credited in the SQN accident analysis and do not provide any additional safety margin. The results of accident analyses remain unchanged by this request. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, 6A West Tower, Knoxville, TN 37902.

NRC Acting Branch Chief: Brian W. Tindell.

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

Date of amendment request: March 29, 2018. A publicly-available version is in ADAMS under Accession No. ML18102A516.

Description of amendment request: The amendments would revise Technical Specification 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' to change the applicability of when the automatic auxiliary feedwater actuation due to the trip of all main feedwater pumps is required to be operable at Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2.

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

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

Response: No.

The design basis events which impose auxiliary feedwater safety function requirements are loss of all AC [alternating current] power to plant auxiliaries, loss of normal feedwater, steam generator fault in either the feedwater or steam lines, and small break loss of coolant accidents. These design basis event evaluations assume actuation of auxiliary feedwater due to station blackout, low-low steam generator level or a safety injection signal. The anticipatory auxiliary feedwater automatic start signals from the main feedwater pumps are not credited in any design basis accidents and are, therefore, not part of the primary success path for postulated accident mitigation as defined by 10 CFR 50.36(c)(2)(ii), Criterion 3. Modifying MODE 2 Applicability for this function will not impact any previously evaluated design basis accidents.

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

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

Response: No.

This technical specification change allows for an operational allowance during MODE 2 while placing main feedwater pumps in service. This change involves an anticipatory auxiliary feedwater automatic start function that is not credited in the accident analysis. Since this change only affects the conditions at which this automatic start function needs to be operable and does not affect the function that actuates auxiliary feedwater due to loss of offsite power, low-low steam generator level or a safety injection signal, it will not be an initiator to a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

This technical [s]pecification change involves the automatic start of the auxiliary feedwater pumps due to trip of both main feedwater pumps, which is not an assumed start signal for design basis events. This change does not modify any values or limits involved in a safety related function or accident analysis.

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

NRC Branch Chief: Robert J. Pascarelli.

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

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

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

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

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

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: June 8, 2017.

Brief description of amendment: The amendment revised technical specifications (TSs) to reflect previously approved changes made as part of the alternative source term initiative. The amendment revised the surveillance requirements for the control room emergency recirculation and annulus exhaust gas treatment systems, which are consistent with Technical Specification Task Force (TSTF) Traveler TSTF–522, ''Revise Ventilation System Surveillance Requirement to Operate for 10 Hours per Month.'' The amendment also deleted two TS sections related to the fuel handling building and fuel handling building ventilation exhaust system and increased the allowable secondary containment leakage. Lastly, the amendment revised the TS Table of Contents to reflect administrative changes to the titles of TS sections.

Date of issuance: May 16, 2018.

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

Amendment No.: 180. A publicly-available version is in ADAMS under Accession No. ML18110A133; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Facility Operating License No. NPF–58: The amendment revised the Facility Operating License and TSs.

*Date of initial notice in **Federal Register:*** August 1, 2017 (82 FR 35841). The supplemental letter dated January 30, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register.**

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2018.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 24, 2017.

Brief description of amendment: The amendment revised the DAEC Technical Specification (TS) Table 3.3.2.1–1, ''Control Rod Block Instrumentation,'' by relocating certain cycle-specific Minimum Critical Power Ratio values to the DAEC Core Operating Limits Report. The amendment also added a requirement to DAEC TS 5.6.5, ''Core Operating Limits Report.''

Date of issuance: March 7, 2018.

Effective date: As of the date of its issuance and shall be implemented by September 27, 2018. (Note: This Notice of Issuance corrects the ''Effective date'' of Amendment No. 303 originally noticed in the **Federal Register** on March 27, 2018 (83 FR 13153).

Amendment No.: 303. A publicly-available version is in ADAMS under Accession No. ML18011A059; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment. Amendment No. 303 was corrected by letter dated May 7, 2018 (ADAMS Accession No. ML18081A074).

Renewed Facility Operating License No. DPR–49: The amendment revised the Renewed Facility Operating License and TSs.

*Date of initial notice in **Federal Register:*** May 23, 2017 (82 FR 23627).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2018.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 29th day of May, 2018.

For the Nuclear Regulatory Commission.

Gregory F. Suber,

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

[FR Doc. 2018–11843 Filed 6–4–18; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Meeting of the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on APR1400

The ACRS Subcommittee on APR1400 will hold a meeting on June 5, 2018, at 11545 Rockville Pike, Room T–2B1, Rockville, Maryland 20852.

The meeting will be open to public attendance with the exception of portions that may be closed to protect information that is proprietary pursuant to 5 U.S.C. 552b(c)(4). The agenda for the subject meeting shall be as follows:

Tuesday, June 5, 2018, 8:30 a.m. Until 5:00 p.m.

The Subcommittee will review the APR1400 Design Control Document and Safety Evaluation Report with No Open Items, Chapter 17 (Quality Assurance & Reliability Assurance), Chapter 19.1 (Probabilistic Risk Assessment), and Chapter 19.2 (Severe Accident Evaluation).

The Subcommittee will hear presentations by and hold discussions with the NRC staff and Korea Hydro & Nuclear Power Company regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the Full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official (DFO), Christopher Brown (Telephone 301–415–7111 or Email: *Christopher.Brown@nrc.gov*) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Thirty-five hard copies of each presentation or handout should be provided to the DFO thirty minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the DFO one day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the DFO with a CD containing each presentation at least thirty minutes before the meeting. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 4, 2017 (82 FR 46312). The bridgeline number for this meeting is 866–822–3032, passcode 8272423#.

Detailed meeting agendas and meeting transcripts are available on the NRC website at *http://www.nrc.gov/reading-rm/doc-collections/acrs.* Information regarding topics to be discussed, changes to the agenda, whether the meeting has been canceled or rescheduled, and the time allotted to present oral statements can be obtained from the website cited above or by contacting the identified DFO. Moreover, in view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with these references if such rescheduling would result in a major inconvenience.

If attending this meeting, please enter through the One White Flint North Building, 11555 Rockville Pike, Rockville, Maryland 20852. After registering with Security, please contact Ms. Kendra Freeland (Telephone 301–415–6207) to be escorted to the meeting room.

Dated: May 23, 2018.

Mark L. Banks,

Chief, Technical Support Branch, Advisory Committee on Reactor Safeguards.

[FR Doc. 2018–12022 Filed 6–4–18; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS) Meeting of the ACRS Subcommittee on Nuscale; Notice of Meeting

The ACRS Subcommittee on NuScale will hold a meeting on June 6, 2018, at 11545 Rockville Pike, Room T–2B1, Rockville, Maryland 20852.

The meeting will be open to public attendance. The agenda for the subject meeting shall be as follows:

Wednesday, June 6, 2018, 8:30 a.m. Until 12:00 p.m.

The Subcommittee will review the staff's SER with open items for Chapter 8, ''Electrical Systems,'' of the NuScale design certification application. The Subcommittee will hear presentations by and hold discussions with the NRC staff and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the Full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official (DFO), Michael Snodderly (Telephone 301–415–2241 or Email: *Michael.Snodderly@nrc.gov*) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Thirty-five hard copies of each presentation or handout should be provided to the DFO thirty minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the DFO one day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the DFO with a CD containing each presentation at least thirty minutes before the meeting. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 4, 2017 (82 FR 46312). The bridgeline number for this meeting is 866–822–3032, passcode 8272423#.

Detailed meeting agendas and meeting transcripts are available on the NRC website at *http://www.nrc.gov/reading-rm/doc-collections/acrs.* Information regarding topics to be discussed, changes to the agenda, whether the meeting has been canceled or rescheduled, and the time allotted to present oral statements can be obtained from the website cited above or by contacting the identified DFO. Moreover, in view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with these references if such rescheduling would result in a major inconvenience.

If attending this meeting, please enter through the One White Flint North building, 11555 Rockville Pike, Rockville, Maryland. After registering with Security, please contact Mr. Theron Brown (Telephone 301–415–6702 or 301–415–8066) to be escorted to the meeting room.