[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2013-0069]

Biweekly Notice

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

Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 21 to April 3, 2013. The last biweekly notice was published on April 2, 2013 (78 FR 19746).

ADDRESSES: You may access information and comment submissions related to this document, which the NRC possesses and is publicly-available, by searching on http://www.regulations.gov under Docket ID **NRC-2013-0069**. You may submit comments by any of the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket **NRC-2013-0069**. Address questions about NRC dockets to Carol Gallagher; telephone: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.
- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.
- **Fax comments to:** RADB at 301-492-3446.

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

SUPPLEMENTARY INFORMATION:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID **NRC-2013-0069** when contacting the NRC about the availability of information regarding this document. You may access information related to this document, which the NRC possesses and is publicly available, by the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0069**.
- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly available documents online in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public

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

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

B. Submitting Comments

Please include Docket ID **NRC-2013-0069** in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information in comment submissions that you do not want to be publicly disclosed. The NRC posts all comment submissions at http://www.regulations.gov as well as entering the comment submissions into ADAMS, and the NRC does not edit comment submissions to remove identifying or contact information.

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

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

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

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

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

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

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

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

entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the requestor/petitioner seeks to have litigated at the proceeding.

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

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

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

significant hazards consideration, then any hearing held would take place before the issuance of any amendment.

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

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

Information about applying for a digital ID certificate is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in the NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at

http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

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

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with the NRC guidance available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC'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 documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for

and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

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

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

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at http://ehd1.nrc.gov/ehd/, unless excluded

pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. However, a request to intervene will require including information on local residence in order to demonstrate a proximity assertion of interest in the proceeding. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Requests for hearing, petitions for leave to intervene, and motions for leave to file new or amended contentions that are filed after the 60-day deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the following three factors in 10 CFR 2.309(c)(1): (i) the information upon which the filing is based was not previously available; (ii) the information upon which the filing is based is materially different from information previously available; and (iii) the filing has been submitted in a timely fashion based on the availability of the subsequent information.

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

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

Date of amendment request: January 11, 2013.

Description of amendment request: The proposed amendment would revise Fermi 2 Technical Specifications (TS) to incorporate the NRC-approved TSTF-423, Revision 1. The proposed amendment would modify TS to risk-inform requirements regarding selected Required Action end states by incorporating the boiling water reactor (BWR) owner's group (BWROG) approved Topical Report NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants." Additionally, the proposed amendment would modify the TS Required Actions with a Note prohibiting the use of limiting condition for operation (LCO) 3.0.4.a when entering the preferred end state (Mode 3) on startup.

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 allows a change to certain required end states when the TS Completion Times for remaining in power operation will be exceeded. Most of the requested technical specification (TS) changes are to permit an end state of hot shutdown (Mode 3) rather than an end state of cold shutdown (Mode 4) contained in the current TS. The request was limited to: (1) those end states where entry into the shutdown mode is for a short interval, (2) entry is initiated by inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical. Risk insights from both the qualitative and quantitative risk assessments were used in specific TS assessments. Such assessments are documented in Section 6 of topical report NEDC-32988-A, Revision 2, "Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR Plants." They provide an integrated

discussion of deterministic and probabilistic issues, focusing on specific TSs, which are used to support the proposed TS end state and associated restrictions. The NRC staff finds that the risk insights support the conclusions of the specific TS assessments. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident after adopting TSTF-423 are no different than the consequences of an accident prior to adopting TSTF-423. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns.

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

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

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). If risk is assessed and managed, allowing a change to certain required end states when the TS Completion Times for remaining in power operation are exceeded (i.e., entry into hot shutdown rather than cold shutdown to repair equipment) will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change and the commitment by the licensee to adhere to the guidance in TSTF-IG-05-02, "Implementation Guidance for TSTF-423, Revision 1, ‘Technical Specifications End States, NEDC-32988-A," will further minimize possible concerns.

Thus, based on the above, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

Response: No.

The proposed change allows, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed. The BWROG’s risk assessment approach is comprehensive and follows NRC staff guidance as documented in Regulatory Guides (RG) 1.174 and 1.177. In addition, the analyses show that the criteria of the three-tiered approach for allowing TS changes are met. The risk

impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A risk assessment was performed to justify the proposed TS changes. The net change to the margin of safety is insignificant.

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: Bruce R. Masters, DTE Energy, General Counsel - Regulatory, 688 WCB, One Energy Plaza, Detroit, MI 48226-1279.

NRC Branch Chief: Robert D. Carlson.

Duke Energy Carolinas, LLC, Docket Nos. 50-269, 50-270, and 50-287, Oconee Nuclear Station, Units 1, 2, and 3 (ONS1, ONS2, and ONS3), Oconee County, South Carolina

Date of amendment request: October 30, 2012.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) to allow operation of a reverse osmosis system during normal plant operation to purify the water in the borated water storage tanks and the spent fuel pools.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration, which is 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 requests NRC's approval of design features and controls that will be used to ensure that periodic limited operation of a Reverse Osmosis (RO) System during Unit operation does not significantly impact the Borated Water Storage Tank (BWST) or Spent Fuel Pool (SFP) function or other plant equipment. The proposed change also requests NRC to approve proposed Technical Specification (TS) requirements that will impose operating restrictions and isolation requirements on the RO System. Duke Energy evaluated the effect of potential failures, identified precautionary measures that must be taken before and during RO System operation, and identified specific required operator actions to protect affected structures, systems, and components (SSCs) important to safety.

The new high energy piping and non-seismic piping being installed for the RO System is non-QA1 and is postulated to fail and cause an Auxiliary Building flood. Duke Energy determined that adequate time is available to isolate the flood source (BWST or SFP) prior to affecting SSCs important to safety.

The existing Auxiliary Building Flood evaluation postulates a single break in the non-seismic piping occurring in a seismic event. The addition of the RO System will not increase the probability of a seismic event. The existing postulated source of the pipe break in the Auxiliary Building is due to the piping not being seismically designed. The new RO System piping is considered a potential source of a single pipe break for the same reason. The new non-seismic RO System piping is of similar quality as the existing non-seismic piping and is no more likely to fail than the existing piping. As such, the addition of new non-seismic piping does not significantly increase the probability of occurrence of an Auxiliary Building flood due to a single pipe break. An Auxiliary Building flood due to a non-seismic RO System pipe break does not increase the consequences of the flood since the new non-seismic pipe break is bounded by the Auxiliary Building flood caused by existing non-seismic pipe breaks.

Procedural controls will ensure that the boron concentration does not go below the TS limit as a result of water returned from the RO System with lower boron concentration. Thus, no adverse effects from decreased boron concentration will occur.

The RO System takes suction from the top of the SFP to protect SFP inventory. Plant procedures will prohibit the use of the RO System for the Units 1 & 2 SFP during the time period directly after an outage that requires the Units 1 & 2 SFP level to be maintained higher than the TS Limiting Condition for Operation (LCO) 3.7.11 level requirement. The higher level is required to support TS LCO 3.10.1 requirements for Standby Shutdown Facility (SSF) Reactor Coolant (RC) Makeup System operability (due to the additional decay heat from the recently offloaded spent fuel). Plant procedures will also specify the siphon be broken

during this time period so the SFP water above the RO suction point cannot be siphoned off if the RO piping breaks. The proposed change does not impact the fuel assemblies, the movement of fuel, or the movement of fuel shipping casks. The SFP boron concentration, level, and temperature limits will not be outside of required parameters due to restrictions/requirements on the system's operation. In addition, the proposed new TS will require the siphon be broken during movement of irradiated fuel assemblies in the SFP or movement of cask over the SFP. Therefore, RO System operation cannot occur during these activities, effectively eliminating a Fuel Handling Accidents (FHA) from occurring while the RO System is in operation.

The BWST is used for mitigation of Steam Generator Tube Rupture (SGTR), Main Steam Line Break (MSLB), and Loss of Coolant Accidents (LOCAs). The SGTR and MSLB are bounded by the small break (SBLOCA) analyses with respect to the performance requirements for the High Pressure Injection (HPI) System. In the normal mode of Unit operation, the BWST is not an accident initiator. The SFP is evaluated to maintain acceptable criticality margin for all abnormal and accident conditions including FHAs and cask drop accidents. Both the BWST and SFP are specified by TS requirements to have minimum levels/volumes and boron concentrations. The BWST also has TS requirements for temperature. Prior to RO System operation, procedures will require the minimum required initial boron concentration and initial level/volume to be adjusted. Additionally, they will require the RO System to be operated for a specified maximum time period before readjusting volume and boron concentration prior to another RO session. This ensures that the TS specified boron concentration and level/volume limits for both the SFP and the BWST are not exceeded during RO System operation. Thus, the design functions of the BWST and the SFP will continue to be met during RO System operation.

Since the BWST and SFP will still have TS boron concentration and level/volume requirements and the RO System will be isolated prior to increasing radiation levels preventing access to the isolation valve, the mitigation of a LOCA or FHA does not result in an increase in dose consequence. Since the design basis LOCA analysis for Oconee assumes 5 gpm back-leakage from the Reactor Building sump to the BWST, the Emergency Operating Procedure will require the RO System to be isolated from the BWST prior to switch over to the recirculation phase. The proposed TS will require the RO system to be isolated (by breaking the siphon) from the SFPs during fuel handling activities and will require the seismic boundary valve between the BWST and RO System to be OPERABLE in MODES 1, 2, 3, and 4.

The additional controls imposed by the proposed Technical Specifications (TSs) will provide additional assurance that isolation valves and operating

restrictions credited to eliminate the need to analyze new release pathways introduced by the RO system will be in place.

Therefore, installation and operation of the RO System during Unit operation and the proposed TS imposing operating restrictions do not significantly increase the probability or consequences of any 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 RO System adds non-seismic piping in the Auxiliary Building. However, the break of a single non-seismic pipe in the Auxiliary Building has already been postulated as an event in the licensing basis. The RO System also does not create the possibility of a seismic event concurrent with a LOCA since a seismic event is a natural phenomena event. The RO System does not adversely affect the Reactor Coolant System pressure boundary. The suction to the RO System, when using the system for BWST purification, contains a normally closed manual seismic boundary valve so the seismic design criteria is met for separation of seismic/non-seismic piping boundaries.

Duke Energy also evaluated potential releases of radioactive liquid to the environment due to RO System piping failures. Design features, controls imposed by the proposed TS, and procedural controls will preclude release of radioactive materials outside the Auxiliary Building by ensuring the RO System will be isolated when required.

The SFP suction line is designed such that the SFP water level will not go below TS required levels, thus the fuel assemblies will have the TS required water level over them. Procedural controls will restrict the use of the RO System and require breaking vacuum on the Units 1 & 2 SFP suction line when the SSF conditions require the SFP level be raised to support SSF RC Makeup System operability. Thus, the SFP water level will not be reduced below required water levels for these conditions. RO System operating restrictions will prevent reducing the SFP boron concentration below TS limits.

Since the BWST and SFP will still have TS boron concentration and level/volume requirements and the RO System will be isolated prior to increasing radiation levels preventing access to the isolation valve, the mitigation of a LOCA or FHA does not result in an increase in dose consequence. Since the design basis LOCA analysis for Oconee assumes 5 gpm back-leakage from the Reactor Building sump to the BWST, the Emergency Operating Procedure will require the RO System to be isolated from the BWST prior to switch over to the recirculation

phase. The proposed TS will require the RO system to be isolated (by breaking the siphon) from the SFPs prior to movement of irradiated fuel assemblies in the SFP or movement of cask over the SFP and will require the seismic boundary valve between the BWST and RO System to be OPERABLE in MODES 1, 2, 3, and 4.

The additional controls imposed by the proposed TSs will provide additional assurance that isolation valves and operating restrictions credited to eliminate the need to analyze new release pathways introduced by the RO system will be in place.

Therefore, operation of the RO System during Unit operation will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

Response: No.

The RO System adds non-seismic piping in the Auxiliary Building. Duke Energy evaluated the impact of RO System operation on SSCs important to safety and determined that the proposed TS controls and procedural controls will ensure that TS limits for SFP and BWST volume, temperature, and boron concentration will continue to be met during RO operation. For the BWST, these controls will ensure the TS minimum BWST boron concentration and level are available to mitigate the consequences of a small break LOCA or a large break LOCA. For the SFP, these controls ensure the assumptions of the fuel handling and cask drop accident analyses are preserved. Additionally, the failure of non-seismic RO System piping will not significantly impact SSCs important to safety. Oconee's licensing basis does not assume a design basis event occurs simultaneously with a seismic event. The proposed change does not significantly impact the condition or performance of SSCs relied upon for accident mitigation. This change does not alter the existing TS allowable values or analytical limits. The existing operating margin between Unit conditions and actual Unit setpoints is not significantly reduced due to these changes. The assumptions and results in any safety analyses are not impacted. Therefore, operation of the RO System during Unit operation 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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202-1802.

NRC Branch Chief: Robert J. Pascarelli.

Duke Energy Carolinas, LLC, Docket Nos. 50-269, 50-270, and 50-287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: February 22, 2013.

Description of amendment request: The proposed amendments would revise the Technical Specification curves for pressure and temperature limits on the reactor coolant system, and limits on heatup and cooldown rates.

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

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

   Response: No.

   The proposed amendment replaces the current Oconee Nuclear Station (ONS) Units 1, 2, and 3 pressure/temperature (P-T) limit curves applicable to 33 effective full power years (EFPY) in Technical Specification (TS) 3.4.3 with new P-T limit curves applicable to 54 EFPY. The proposed changes also revise the Reactor Coolant System heatup and cooldown rates and allowable reactor coolant pump combinations of TS Tables 3.4.3-1 and 3.4.3-2. The pressure-temperature (P-T) limit curves in the TSs were conservatively generated in accordance with fracture toughness requirements of ASME Code Section XI, Appendix G, and the minimum pressure and temperature requirements as listed in Table 1 of 10 CFR Part 50, Appendix G. The proposed changes do not impact the capability of the reactor coolant pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.).

Therefore, the proposed changes do not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed P-T limits, heatup and cooldown rates and allowable operating reactor coolant pump combinations are not considered to be an initiator or contributor to any accident analysis addressed in the ONS Updated Final Safety Analyses Report (UFSAR).

The proposed changes will not impact assumptions and conditions previously used in the radiological consequence evaluations nor affect the mitigation of these consequences due to an accident described in the UFSAR. Also, the proposed changes will not impact a plant system such that previously analyzed SSCs might be more likely to fail. The initiating conditions and assumptions for accidents described in the UFSAR remain as analyzed.

Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

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

Response: No.

The requirements for P-T limit curves have been in place since the beginning of plant operation. The revised curves are based on a later edition to Section XI of the ASME Code that incorporates current industry standards for P-T curves. The revised curves are based on reactor vessel irradiation damage predictions using Regulatory Guide 1.99 methodology. No new failure modes are identified nor are any SSCs required to be operated outside the design bases.

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

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

Response: No.

The proposed P-T curves continue to maintain the safety margins of 10 CFR Part 50, Appendix G, by defining the limits of operation which prevent non-ductile failure of the reactor pressure vessel. Analyses have demonstrated that the fracture toughness requirements are satisfied and that conservative operating restrictions are maintained for the purpose of

> low temperature overpressure protection. The P-T limit curves provide assurance that the RCS pressure boundary will behave in a ductile manner and that the probability of a rapidly propagating fracture is minimized.
>
> Therefore, this request 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: Lara S. Nichols, Deputy General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202-1802.

NRC Branch Chief: Robert J. Pascarelli.

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

Docket Nos. STN 50-454 and STN 50-455, Byron Station, Units 1 and 2, Ogle County, Illinois

Date of amendment request: December 21, 2012.

Description of amendment request: The proposed amendment would Revise Technical Specifications (TS) 3.3.6, "Containment Ventilation Isolation Instrumentation." Specifically, this amendment request proposes to revise Footnote (b) of TS Table 3.3.6-1, "Containment Ventilation Isolation Instrumentation," which specifies the "Containment Radiation – High" trip setpoint for two containment area radiation monitors (i.e., 1(2) RE-AR011 and 1(2) RE-AR012). The proposed changes would revise the "Containment Radiation- High" trip setpoint from the current, overly conservative value (i.e., a submersion dose rate of less than or equal to 10 mRhr

in the containment building), to less than or equal to 2 times the containment building background radiation reading at rated thermal power, which is consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Upon reaching the "Containment Radiation - High" setpoint, these area radiation monitors provide an isolation signal to the containment normal purge, mini purge and post-LOCA (Loss of Coolant Accident) systems' containment isolation valves.

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

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

   Response: No.

   The containment ventilation isolation radiation monitors serve two primary functions, they:

   a. act as backup to the SI [safety injection] signal to ensure closing of the purge valves; and
   b. are the primary means for automatically isolating containment in the event of a fuel handling accident in containment.

   Upon sensing a high radiation condition in containment, these area radiation monitors provide an isolation signal to the containment normal purge, mini purge and post- LOCA systems containment isolation valves (i.e., a containment ventilation isolation signal).

   The accidents that could potentially be impacted by the proposed change were evaluated; specifically the Loss of Coolant Accident (LOCA), Control Rod Ejection Accident (CREA) and Fuel Handling Accident (FHA) in Containment. The proposed change has no impact on probability of these accidents occurring as the subject containment radiation area monitors detect a high radiation condition resulting from these accidents. The radiation monitors do not initiate any accidents or transients. Changing the "Containment Radiation – High" trip setpoint from "≤ 10 mR/hr in the containment building," to "≤ 2 times the containment building background radiation reading at rated thermal power" only affects the point (i.e., the radiation level in containment) at which a containment ventilation isolation signal would be generated. The requested change

does not involve any physical plant modifications or operational changes that could adversely affect system reliability or performance of the radiation monitors, or that could affect the probability of operator error.

The requested change does not affect any postulated accident precursors and therefore, will not affect the probability of an accident previously evaluated.

The proposed change was evaluated to determine the impact on the dose consequences of the LOCA, CREA, or FHA in containment. The evaluation assumed a containment purge was in progress at the onset of the subject accidents and showed that the proposed change in the containment radiation monitors' setpoint had no effect on the purge valve isolation time. With regard to the LOCA and CREA, the safety analysis assumes a prompt purge valve isolation time (i.e., approximately 60 seconds) that significantly bounds the radiation monitor sensing and response time, and actual valve design closure time (i.e., a total of approximately 7 seconds). The radiation monitor setpoint is not considered in the safety analysis and any dose contribution associated with the containment purge, due to the proposed change in setpoint, was shown to be unaffected; therefore, the proposed change has no impact on the already insignificant dose contribution attributed to a containment purge during an accident of less than one mrem.

The dose consequences associated with the FHA in containment are also not impacted by the proposed change in containment radiation monitor setpoint. The existing dose consequences resulting from a FHA with moving non-RECENTLY IRRADIATED FUEL (i.e., fuel moved more than 48 hours after reactor shutdown) conservatively assume the containment purge valves remain open throughout the event; therefore, a change in the isolation setpoint does not impact the results of this analysis. With regard to movement of RECENTLY IRRADIATED FUEL (i.e., fuel moved less then 48 hours after reactor shutdown), EGC's [Exelon Generation Company] proposal deletes TS LCO [limiting condition for operation] 3.9.4.c.2 which allowed the containment purge valves to be open provided the containment radiation isolation system is OPERABLE. Deletion of TS LCO 3.9.4.c.2 ensures that the containment purge valves are in the closed position when moving RECENTLY IRRADIATED FUEL, thus removing dependence on the containment radiation isolation system and associated radiation monitor setpoint from the FHA dose consequences.

The four other additional TS changes associated with the deletion of LCO 3.9.4, Item c.2, proposed for consistency (i.e., deleting a NOTE regarding MODE applicability, deleting a CONDITION related only to LCO 3.9.4.c.2, deleting a footnote regarding MODE applicability; and deleting two surveillances related to LCO 3.9.4.c.2), also have no affect on either the probability or consequences of an accident previously evaluated.

Based on the above discussion, 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 changes do not result in a change to the design of the Containment Ventilation Isolation System or the manner in which the system operates or provides plant protection. The containment radiation monitors will sense radiation levels in the same way and will respond in the same manner when the setpoint is exceeded. The change in the "Containment Radiation – High" setpoint does not create a new failure mode for the associated containment radiation monitors or for any other plant equipment. The deletion of LCO 3.9.4, Item c.2, in support of the setpoint change during refueling operations, is more conservative than the current allowances and actually eliminates a potential failure mode for the assumed open containment ventilation isolation valves as the proposed deletion of LCO 3.9.4, Item c.2 would require the valves to be closed prior to moving RECENTLY IRRADIATED FUEL.

The changes do not result in the creation of any new accident precursors, the creation of any changes to the existing accident scenarios, nor do they create any new or different accident scenarios. Subsequently, the accidents defined in the UFSAR [updated final safety analysis report] continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

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

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

Response: No.

The analysis methodologies used in the subject safety analyses are not modified as a result of the proposed TS changes to the "Containment Radiation – High" trip setpoint or the deletion of LCO 3.9.4, Item c.2, or any of the other four associated TS changes. Although the "Containment Radiation – High" trip setpoint is being increased, the increase in response time to a high radiation condition in containment, when compared to the current setpoint, is negligible due to the projected prompt rise in containment radiation level upon initiation of a LOCA. The dose consequences and resultant margin of safety to the regulatory acceptance limits, due to revising the "Containment Radiation – High"

setpoint to ≤ 2 times the containment building background radiation reading at rated thermal power, was shown to be unaffected for normal at-power containment releases; have a negligible impact on the associated LOCA and CREA accident dose consequences; and have no impact on the FHA when moving RECENTLY IRRADIATED FUEL. Therefore, the proposed changes do not impact any analysis margins.

The proposed changes do not alter the manner in which the safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The current safety analyses remain bounding since their conclusions are not affected by the proposed changes. The safety systems credited in the safety analyses will continue to be available to perform their mitigation functions. All protection signals credited as the primary or secondary accident mitigating functions, and all operator actions credited in the accident analyses remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis.

Based on the above information, the proposed change does not result in a significant reduction in the margin of safety.

Based on the above evaluation, EGC concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and, accordingly, a finding of no significant hazards consideration is justified.

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: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Acting Branch Chief: Jeremy S. Bowen.

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

Date of amendment request: January 29, 2013.

Description of amendment request: The license amendment request proposes to remove completed and satisfied license conditions and to correct inadvertent errors and incorrect references.

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 amendments do not change or modify the fuel, fuel handling processes, fuel storage racks, number of fuel assemblies that may be stored in the spent fuel pool (SFP), decay heat generation rate, or the spent fuel pool cooling and cleanup system. The proposed amendments only limit crediting of burnable absorbers in the spent fuel pool to Integrated Fuel Burnable Absorber (IFBA) rods that were specifically addressed in the currently approved criticality analysis ([Westinghouse Commercial Atomic Power report] WCAP-1 7094-P, Revision 3). The removal of the phrase “or an equivalent amount of another burnable absorber” eliminates the possibility of crediting a burnable absorber other than IFBA for storage of spent fuel assemblies in the spent fuel pool without prior NRC’s approval. The deletion of the license condition associated with the Boraflex Remedy is editorial as it is no longer applicable. The proposed amendments do not affect the ability of the BAST [boric acid storage tank] to perform its function or the ability of the CREVS [control room emergency ventilation system] to perform its function. These latter proposed TS [technical specification] changes correct inadvertent errors and are consistent with the stated intent of original license submittals or delete license conditions that are no longer applicable or that have been fully satisfied.

   The proposed amendments do not cause any physical change to the existing spent fuel storage configuration, fuel makeup, RCS [reactor coolant system] pressure boundary, reactor containment, or plant systems. The proposed amendments do not affect any precursors to any accident previously evaluated or do not affect any known mitigation equipment or strategies.

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

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

   Response: No.

   The proposed amendments do not change or modify the fuel, fuel handling processes, fuel racks, number of fuel assemblies that may be stored in the pool, decay heat generation rate, or the spent fuel pool cooling and cleanup system. The proposed amendments do not result in any changes to spent fuel or to fuel storage configurations. The removal of the phrase "or an equivalent amount of another burnable absorber" eliminates the possibility of crediting a burnable absorber other than IFBA for storage of spent fuel assemblies in the spent fuel pool without prior NRC approval. The proposed amendments do not affect the ability of the BAST to perform its function or the ability of the CREVS to perform its function. These latter proposed TS changes correct inadvertent errors and are consistent with the stated intent of the original license submittals, delete license conditions that are no longer applicable or have been fully satisfied.

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

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

   Response: No.

   The proposed amendments do not change or modify the fuel, fuel handling processes, fuel racks, number of fuel assemblies that may be stored in the pool, decay heat generation rate, or the spent fuel pool cooling and cleanup system. Therefore, the proposed amendments have no impact to the existing margin of safety for subcriticality required by 10 CFR 50.68(b)(4). The other proposed OL [operating license] & TS changes correct inadvertent errors and are consistent with the stated intent of the original license submittals or delete license conditions that are no longer applicable or have been fully satisfied.

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

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

Attorney for licensee: James Petro, Managing Attorney - Nuclear, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408-0420.

NRC Branch Chief: Jessie F. Quichocho.

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

Date of amendment request: June 25, 2012.

Description of amendment request: The amendment would revise the description of the Fuel Handling Accident (FHA) in Section XIV-6.4 of the Cooper Nuclear Station (CNS) Updated Safety Analysis Report (USAR). The revised USAR FHA description is based on changes to the Design Basis Accident FHA dose calculation, to reflect a 24-month fuel cycle source term using a Global Nuclear Fuels (GNF) 10 x 10 fuel array, reduce the bounding Radial Peaking Factor, and revise the total effective dose equivalent (TEDE) contribution to consider the shine contribution from external sources.

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 analyses changes described by this proposed change to the USAR are not initiators to events, and, therefore, do not involve the probability of an accident. The changes to the FHA calculation for radiological dose following a FHA incorporate the following:

   – accounts for the increase to the source term owing to the use of Global Nuclear Fuels (GNF) 10 x 10 fuel exposed over a 24-month fuel cycle,

- reduces the Radial Peaking Factor from 2.0 to 1.95, and
- uses a calculated Control Room shine contribution that is added to the FHA dose consequences.

The NRC computer code RADTRAD Version 3.03 is used for the offsite and Control Room dose calculation. The RADTRAD code was approved for use with the CNS FHA alternative source term (AST) dose calculation in License Amendment 222.

Because the analysis affected by the changes are not considered to be an initiator to any previously analyzed accident, these changes cannot increase the probability of any previously evaluated accident. Therefore, these changes do not increase the probability of occurrence of an accident evaluated previously in the USAR.

The changes in FHA dose consequences to the Control Room occupant resulting from the 24-month cycle/GNF 10 x 10 source term (without crediting the offset by a reduced Radial Peaking Factor), results in more than a minimal increase in the consequences of an accident previously evaluated in the USAR, as stated in 10 CFR 50.59(c)(2)(iii). However, the resultant dose remains well within the regulatory limits of 10 CFR 50.67. When the reduced Radial Peaking Factor is applied, the dose consequences are minor. Therefore, this change does not significantly increase the consequences of an accident previously evaluated in the USAR.

In summary, 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 change does not involve initiators to any events in the USAR, nor does the activity create the possibility for any new accidents. Rather, this change is a result of the evaluation of the most limiting FHA, which can occur at CNS. The changes to the FHA calculation for radiological dose following a FHA incorporate the following:

- accounts for the increase to the source term owing to the use of GNF 10 x 10 fuel exposed over a 24-month fuel cycle,
- reduces the Radial Peaking Factor from 2.0 to 1.95, and
- uses a calculated Control Room shine contribution that is added to the FHA dose consequences.

The RADTRAD code accommodates the use of GNF 10 x 10 fue0l exposed over a 24-month fuel cycle in calculating the FHA dose

consequences. The reduction in Radial Peaking Factor remains bounding over the CNS core design. The calculated Control Room shine contribution replaces the previously approved qualitative assessment. The proposed change does not introduce any new modes of plant operation and does not involve physical modifications to the plant. As a result, no new failure modes are being introduced.

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

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

Response: No.

The dose consequences are calculated in accordance with the regulatory guidance found in RG 1.183. The RADTRAD code was used, as approved for application at CNS with License Amendment 222. With the reduced Radial Peaking Factor applied to the GNF 10 x 10 fuel that has been exposed over a 24-month fuel cycle, the dose consequences are minor. The calculated shine contribution being added to the total Control Room occupant FHA dose results are less than the previous qualitative assessment results that are being replaced. Accordingly, the safety margins to the regulatory dose limits are preserved.

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: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602-0499.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: November 13, 2012.

Description of amendment request: The proposed amendment would revise Renewed Facility Operating License (RFOL) Condition C.12 to clarify that the programs and activities, identified in Appendix A of NUREG-1955 and the Updated Final Safety Analysis Report (UFSAR) supplement are to be completed before the period of extended 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.

   The amendment does not significantly increase the probability of an accident since it does not involve a change to any plant equipment that initiates a plant accident. The change clarifies RFOLC [RFOL Condition] C.12. The license conditions deal with administrative controls over information contained in the Updated Final Safety Analysis Repo[r]t (UFSAR) supplement. The proposed changes are administrative and the license conditions are not an initiator or mitigator of any previously evaluated accidents.

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

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

   Response: No.

   The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated since it does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. The license conditions deal with administrative controls over information contained in the UFSAR supplement. No new or different types of equipment will be installed and the basic operation of installed equipment is unchanged.

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

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

Response: No.

The proposed amendment does not affect design codes or design margins. The change that clarifies RFOLC C.12 is administrative in nature and does not have the ability to affect analyzed safety margins.

Therefore, operation of DAEC in accordance with the proposed amendment will not involve a significant reduction in the margin to safety.

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

Attorney for licensee: Mr. James Petro, P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Robert D. Carlson.

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

Date of amendment request: December 21, 2012.

Description of amendment request: The proposed amendment would modify the current DAEC Technical Specifications (TS) requirement to operate the Standby Gas Treatment System for 10 hours on a frequency specified in the Surveillance Frequency Control Program in accordance with TSTF-522, Revision 0, "Revise Ventilation System Surveillance Requirements to Operate for 10 hours per Month."

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 replaces an existing Surveillance Requirement to operate the SGT System equipped with electric heaters for a continuous 10 hour period with a requirement to operate the SGT System for 15 continuous minutes without the heaters operating. In addition, the electrical heater output test in the VFTP (Specification 5.5.7.e) is proposed to be deleted and a corresponding change in the charcoal filter testing (Specification 5.5.7.c) be made to require the testing be conducted at a humidity of at least 95% RH, which is more stringent than the current testing requirement of 70% RH.

   These systems are not accident initiators and therefore, these changes do not involve a significant increase in the probability of an accident. The proposed system and filter testing changes are consistent with current regulatory guidance for these systems and will continue to assure that these systems perform their design function which may include mitigating accidents. Thus the change does not involve a significant increase in the consequences of an accident.

   Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change replaces an existing Surveillance Requirement to operate the SGT System equipped with electric heaters for a continuous 10 hour period with a requirement to operate the systems for 15 continuous minutes without the heaters operating. In addition, the electrical heater output test in the VFTP (Specification 5.5.7.e) is proposed to be deleted and a corresponding change in the charcoal filter testing (Specification 5.5.7.c) be made to require the testing be conducted at a humidity of at least 95% RH, which is more stringent than the current testing requirement of 70% RH.

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

Therefore, it is concluded that this 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 replaces an existing Surveillance Requirement to operate the SGT System equipped with electric heaters for a continuous 10 hour period with a requirement to operate the systems for 15 continuous minutes without heaters operating. In addition, the electrical heater output test in the VFTP is proposed to be deleted and a corresponding change in the charcoal filter testing be made to require the testing be conducted at a humidity of at least 95% RH, which is more stringent than the current testing requirement of 70% RH.

The proposed increase to 95% RH in the required testing of the charcoal filters compensates for the function of the heaters, which was to reduce the humidity of the incoming air to below the currently-specified value of 70% RH for the charcoal. The proposed change is consistent with regulatory guidance and continues to ensure that the performance of the charcoal filters is acceptable.

Therefore, it is concluded that this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. James Petro, P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Robert D. Carlson.

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

Date of amendment request: April 20, 2012.

Description of amendment request: The proposed amendment would revise the TS 3.1.7 to approve the use of an alternative method, other than the current method of the use of movable incore detectors system, to monitor the position of control rod or shutdown rod, in the event of a malfunction of the microprocessor rod position indication (MRPI) system. The use of this alternative method would reduce the required frequency of flux mapping using the movable incore detector system to determine the position of the control or shutdown rod position that is not being indicated. This will reduce the wear on the movable incore detector system that is also used to complete other required TS surveillances

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 provides an alternative method for verifying rod position of one rod. The proposed change meets the intent of the current specification in that it ensures verification of position of the rod once every 8 hours. The proposed change provides only an alternative method of monitoring rod position and does not change the assumptions or results of any previously evaluated accident.

   Therefore, operation of the facility in accordance with the proposed amendment would 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 provides only an alternative method of determining the position of one rod. No new accident initiators are introduced by the proposed alternative manner of performing rod position verification. The proposed change does not affect the reactor protection system. Hence, no new failure modes are created that would cause a new or different kind of accidents from any accident previously evaluated.

Therefore, operation of the facility in accordance with the proposed amendments would 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 basis of TS 3.1.7 states that the operability of the rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. The proposed change does not alter the requirement to determine rod position but provides an alternative method for determining the position of the affected rod. As a result, the initial conditions of the accident analysis are preserved and the consequences of previously analyzed accidents are unaffected.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety

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

Attorney for licensee: Carey Fleming, Sr. Counsel - Nuclear Generation, Constellation Group, LLC, 750 East Pratt Street, 17 Floor, Baltimore, MD 21202.

NRC Acting Branch Chief: Sean Meighan.

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

Date of amendment request: March 26, 2013.

Description of amendment request: The proposed change would amend Combined License Nos.: NPF-93 and NPF-94 for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 by departing from the plant-specific design control document Tier 2* material contained within the Updated Safety Analysis Report (UFSAR) by revising the structural criteria code for anchoring of reinforcement bar within the nuclear island walls.

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

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

   Response: No.

   The design functions of the nuclear island structures are to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in the nuclear island. The nuclear island structures are structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29.

   The change of the requirements for anchoring headed reinforcement does not have an adverse impact on the response of the nuclear island structures to safe shutdown earthquake ground motions or loads due to anticipated transients or postulated accident conditions. The change of the requirements for anchoring headed reinforcement does not impact the support, design, or operation of mechanical and fluid systems. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor does the change described create any new accident precursors.

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

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

Response: No.

The proposed change is to provide the requirements for anchoring nuclear island headed reinforcement. The proposed change does not change the design of the nuclear island structures except to a limited extent to redistribute the shear reinforcement in the walls of the nuclear island. The proposed change does not impact the support, design, or operation of mechanical or fluid systems. The proposed change does not result in a new failure mechanism for the nuclear island structures or new accident precursors. As a result, the design functions of the nuclear island structures and the seismic Category I mechanical and electrical equipment located in the nuclear island are not adversely affected by the proposed change.

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

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

Response: No.

No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed change, thus, no margin of safety is reduced. The limited application of alternative criteria for headed reinforcement does not reduce the margin of safety.

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

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

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

NRC Branch Chief: Lawrence Burkhart, Acting.

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

Date of amendment request: March 20, 2013.

Description of amendment request: The proposed change would amend Combined Licenses Nos.: NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 by departing from the plant-specific design control document Tier 2* material contained within the Updated Safety Analysis Report (UFSAR) by revising the structural criteria code for anchoring of reinforcement bar within the nuclear island walls.

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

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

   Response: No.

   The design functions of the nuclear island structures are to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in the nuclear island. The nuclear island structures are structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29.

   The change of the requirements for anchoring headed reinforcement does not have an adverse impact on the response of the nuclear island structures to safe shutdown earthquake ground motions or loads due to anticipated transients or postulated accident conditions. The change of the requirements for anchoring headed reinforcement does not impact the support, design, or operation of mechanical and fluid systems. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor does the change described create any new accident precursors.

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

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

   Response: No.

   The proposed change is to provide the requirements for anchoring nuclear island headed reinforcement. The proposed change does not change the design of the nuclear island structures except to a limited extent to redistribute the shear reinforcement in the walls of the nuclear island. The proposed change does not impact the support, design, or operation of mechanical or fluid systems. The proposed change does not result in a new failure mechanism for the nuclear island structures or new accident precursors. As a result, the design functions of the nuclear island structures and the seismic Category I mechanical and electrical equipment located in the nuclear island are not adversely affected by the proposed change.

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

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

   Response: No.

   No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed change, thus, no margin of safety is reduced. The limited application of alternative criteria for headed reinforcement does not reduce the margin of safety.

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

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

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

NRC Acting Branch Chief: Lawrence Burkhart.

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

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

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

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

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

Dominion Nuclear Connecticut, Inc., Docket No. 50-336, Millstone Power Station, Unit 2, New London County, Connecticut

Date of amendment request: December 17, 2012, as supplemented by January 31, 2013.

Description of amendment request: The amendment revised the Millstone Power Station, Unit 2 (MPS2) Technical Specification (TS) Surveillance Requirement 4.4.3.2 to remove the requirement to perform the quarterly surveillance for a pressurizer power-operated relief valve (PORV) block valve that is being maintained closed in accordance with TS 3.4.3 Action a. The proposed change is consistent with the requirements of the Standard Technical Specification - Combustion Engineering Plants (NUREG-1432, Revision 4).

Date of issuance: March 26, 2013.

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

Amendment No.: 314.

Renewed Facility Operating License No. DPR-65: Amendment revised the License and Technical Specifications.

Date of initial notice in *Federal Register*: January 22, 2012 (78 FR 4472). The supplemental letter dated January 31, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 2013.

No significant hazards consideration comments received: No.

Entergy Gulf States Louisiana, LLC, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: December 8, 2011, as supplemented by letters dated April 11, May 2, and September 5, 2012, and January 9 and March 8, 2013.

Brief description of amendment: The amendment revised Surveillance Requirement (SR) 3.3.8.1.3 (calibration of loss of power instrumentation) to extend the frequency of the SR from 18 to 24 months, and revised certain Allowable Values in TS 3.3.8.1, "Loss of Power Instrumentation."

Date of issuance: March 29, 2013.

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

Amendment No.: 179.

Facility Operating License No. NPF-47: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: April 17, 2012 (77 FR 22811). The supplemental letters dated April 11, May 2, and September 5, 2012, and January 9 and March 8, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2013.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 6, 2012.

Brief description of amendment: The amendment revised the technical specifications (TS) by adding a new Limiting Condition for Operation (LCO) 3.0.8 associated with the impact of inoperable snubbers. This LCO establishes conditions under which TS systems would remain operable when required snubbers are not capable of providing the related support function. The proposed amendment is consistent with NRC's approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-372, Revision 4, "Addition of LCO 3.0.8, Inoperability of Snubbers."

Date of issuance: March 29, 2013.

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

Amendment No.: 251.

Facility Operating License No. DPR-20: Amendment revised the Technical Specifications.

Date of initial notice in *Federal Register*: November 27, 2012 (77 FR 70841).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2013.

No significant hazards consideration comments received: No.

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

Docket Nos. STN 50-454 and STN 50-455, Byron Station, Units 1 and 2, Ogle County, Illinois

Date of application for amendment: March 22, 2012, as supplemented by letter dated December 3, 2012.

Brief description of amendment: The proposed amendment would modify technical specification (TS) requirements regarding steam generator tube inspections and reporting as described in Technical Specifications Task Force (TSTF)-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," with proposed variations and deviations.

Date of Issuance:. March 25, 2013.

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

Amendment Nos.: 172 and 170.

Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66: The amendments revised the TS and license.

Date of initial notice in *Federal Register*: (77 FR 31660; May 29, 2012). The December 3, 2012, supplement did not increase the scope of the application and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 25, 2013.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 27, 2012, as supplemented on October 15, 2012.

Brief description of amendments: The amendments: (1) adopted a new methodology for preparation of the reactor coolant system pressure-temperature (P-T) limits, (2) relocated the P-T limits in the Technical Specifications (TSs) to a new licensee-controlled document, the Pressure and Temperature Limits Report (PTLR), and (3) modified the TSs to add references to the PTLR.

Date of issuance: April 1, 2013.

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendments Nos.: 286 and 289.

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

Date of initial notice in *Federal Register*: July 3, 2012 (77 FR 39525). The letter dated October 15, 2012, provided clarifying information that did not change the initial proposed no

significant hazards consideration determination or expand the application beyond the scope of the original *Federal Register* notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 1, 2013.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: September 6, 2012, as supplemented by letter dated January 11, 2013.

Brief description of amendments: The amendments revised Technical Specification (TS) 3/4.6.2.3, "Recirculation pH Control System and NaTB Basket Minimum Loading Requirement," to reduce the minimum loading requirement of sodium tetraborate.

Date of issuance: April 2, 2013.

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

Amendment Nos.: 257 and 253.

Renewed Facility Operating License Nos. DPR-31 and DPR-41: Amendments revised the TSs.

Date of initial notice in *Federal Register*: January 25, 2013 (78 FR 5505). The supplement dated January 11, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 2, 2013.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 12, 2012

Brief description of amendments: The amendments revised the Technical Specifications (TSs) to adopt NRC-approved TS Task Force (TSTF) Traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," using the consolidated line item improvement process. Specifically, the amendments revise TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.5.7, "Steam Generator (SG) Program," and TS 5.6.7, "Steam Generator Tube Inspection Report," and include TS Bases changes that summarize and clarify the purpose of the TS.

Date of issuance: March 22, 2013.

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

Amendment Nos.: 320 and 304.

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revise the Operating Licenses and the Technical Specifications.

Date of initial notice in *Federal Register*: December 26, 2012 (77 FR 76080).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 22, 2013.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 9, 2012, as supplemented by letter dated October 31, 2012.

Brief description of amendment: The amendment relocated the Fort Calhoun Station (FCS) Technical Specification (TS) Limiting Condition of Operation (LCO) 2.17, Miscellaneous Radioactive Material Sources, and the associated Surveillance Requirement (SR) 3.13, Radioactive Material Sources Surveillance, from the FCS TSs. NUREG-1432, Revision 3, "Standard Technical Specifications, Combustion Engineering Plants," does not contain a TS or SR for radioactive source surveillance. The operability and surveillance requirements for leak checking of miscellaneous radioactive material sources will be incorporated into the FCS Updated Safety Analysis Report and associated plant procedures.

Date of issuance: March 29, 2013.

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

Amendment No.: 271.

Renewed Facility Operating License No. DPR-40: The amendment revised the facility operating license and the Technical Specifications.

Date of initial notice in *Federal Register*: November 13, 2012 (77 FR 67684). The supplemental letter dated October 31, 2012, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated March 29, 2013.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50-272, Salem Nuclear Generating Station, Unit 1, Salem County, New Jersey

Date of application for amendment: May 8, 2012.

Brief description of amendment: The amendment revised Salem Unit 1 Technical Specification (TS) 6.8.4.i, "Steam Generator (SG) Program," to permanently exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections. In addition, this amendment also revises TS 6.9.1.10, "Steam Generator Tube Inspection Report," to provide permanent reporting requirements that have been previously established on an interim basis. The amendment was submitted pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit."

Date of issuance: March 28, 2013.

Effective date: The license amendment is effective as of the date of issuance and shall be implemented within 60 days.

Amendment No.: 303.

Renewed Facility Operating License No. DPR-70: The amendment revised the facility operating license and the Technical Specifications.

Date of initial notice in *Federal Register*: January 22, 2013 (78 FR 4474).

The Commission' related evaluation of the amendments is contained in a Safety Evaluation dated March 28, 2013.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 5th day of April 2013.

FOR THE NUCLEAR REGULATORY COMMISSION,

/RA/

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