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[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2013-0257]

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 to publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license or combined license, as applicable, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 31, 2013 to November 13, 2013. The last biweekly notice was published on November 12, 2013 (78 FR 67402).

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

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0257**. Address questions about NRC dockets to Carol Gallagher; telephone: 301-287-3422; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individual(s) listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: 3WFN, 06-44M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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-2013-0257** when contacting the NRC about the availability of information regarding this document. You may access publicly-available information related to this action by the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0257**.

- **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. The ADAMS accession number for each document referenced in this notice (if that document is available in ADAMS) is provided the first time that a document is referenced

- **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-2013-0257** 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 that you do not want to be publicly disclosed in your comment submission. The NRC posts all comment submissions at http://www.regulations.gov as well as entering the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment 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 to 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 three factors in 10 CFR 2.309(c)(1)(i)(iii).

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.

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: October 30, 2012, as supplemented on January 21, June 11, September 3, and October 21, 2013.

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. Automatic isolation valves would be installed in the Spent Fuel Pool Cooling (SFPC) system upstream of the Reverse Osmosis (RO) system borated water storage tank suction connections.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), in its supplemental letter dated October 21, 2013, the licensee provided a revised analysis of the issue of no significant hazards consideration, which is presented below:

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

   Response: No.

   The proposed change requests Nuclear Regulatory Commission (NRC) approval of design features and controls that will be used to ensure that unisolating the SFPC Purification System and the Reverse Osmosis (RO) System during Unit operations does not significantly impact the Borated Water Storage Tank (BWST) or other plant equipment and that periodic limited operation of the RO System when aligned to a SFP during Unit operation does not significantly impact the 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 for the SFPC Purification System and the RO System.

   The new high energy piping and non-seismic piping being installed for the RO System is non-QA1 and is postulated to fail. Adequate measures have been provided to isolate the flood source (BWST or SFP) prior to affecting SSCs important to safety.

   The BWST will be automatically isolated prior to going below the TS water volume requirement. For the SFP, the suction to the RO system is above the required TS water level, therefore, the design ensures the required TS water level is maintained.

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 and 2 SFP during the time period directly after an outage that requires the Units 1 and 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 Technical Specification will require the siphon be broken during movement of irradiated fuel assemblies in the SFP or movement of a cask over the SFP. Therefore, RO System operation cannot occur during these activities, effectively eliminating a Fuel Handling Accident (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 operation to be restricted to 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.

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 automatic isolation valves between the BWST and the SFPC Purification System, upstream of the branch line to the RO System branch line, be OPERABLE in MODES 1, 2, 3, and 4. The TS will also require manual valves in branch lines upstream of the SFPC Purification System automatic isolation valves to be closed and meet Inservice Testing (IST) Program leakage requirements.

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

Therefore, allowing the SFPC Purification System and the RO System to be unisolated during Unit operation 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 phenomenon event. The RO System does not adversely affect the Reactor Coolant System pressure boundary.

   Duke Energy also evaluated potential releases of radioactive liquid to the environment. Design features, controls imposed by the proposed Technical Specification, and procedural controls will preclude release of radioactive materials outside the Auxiliary Building by ensuring the SFPC Purification System and 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 and 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 already have TS boron concentration and level/volume requirements and the RO System will be automatically isolated, the mitigation of a LOCA or FHA does not result in an increase in dose consequence. The design basis LOCA analysis for Oconee assumes 5 gpm back-leakage from the Reactor Building sump to the BWST. The automatic isolation valves will isolate on a BWST level prior to swapover to the recirculation phase and prior to going below the actual TS water level. 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 automatic isolation valves 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 will provide additional assurance that isolation valves and operating restrictions credited to eliminate the need to analyze new release pathways (introduced by allowing the SFPC Purification System and the RO system to be unisolated during Unit operation) will be in place.

Therefore, operation of these systems unisolated 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.

   Duke Energy evaluated the impact of allowing the SFPC Purification System and the RO System to be unisolated during Unit 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. 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. 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.

Entergy Nuclear Operations, Inc., Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: June 7, 2013.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 1.1 "Definitions," for Shutdown Margin (SDM), to require calculation of the SDM at a reactor moderator temperature of 68 °F or a higher temperature that is determined to represent the most reactive state throughout the operating cycle of the reactor. This change is needed to address new Boiling Water Reactor (BWR) fuel designs which may be more reactive at shutdown temperatures above 68 °F.

The NRC staff announced the availability of Technical Specifications (TSs) Task Force (TSTF) Traveler TSTF-535, Revision 0, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs." The TSTF-535, Revision 0 provides guidance for plant-specific adoption of changes needed to address BWR fuel designs which may be more reactive at shutdown temperatures greater than 68 °F, using the agency's Consolidated Line Item Improvement Process" (CLIIP). The availability and the model safety evaluation of TSTF-535, Revision 0, was provided under ADAMS Accession No. ML12355A772, and published in the *Federal Register* dated November 19, 2012 (77 FR 69507).

The licensee has reviewed the information provided by the NRC staff in TSTF-535, and the model safety evaluation, as announced in the Federal Register (FR) Notice of availability. The licensee concluded that the justification presented in the FR Notice of availability of TSTF-535, Revision 0 and the model safety evaluation, prepared by the NRC staff, is applicable to the James A. FitzPatrick Nuclear Power Plant and justifies the current request for amendment to TS 1.1, "Definitions" for SDM.

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

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

   Response: No.

   The proposed change revises the definition of SDM. SDM is not an initiator to any accident previously evaluated. Accordingly, the proposed change to the definition of SDM has no effect on the probability of any accident previously evaluated. SDM is an assumption in the analysis of some previously evaluated accidents and inadequate SDM could lead to an increase in consequences for those accidents. However, the proposed change revises the SDM definition to ensure that the correct SDM is determined for all fuel types at all times during the fuel cycle. As a result, the proposed change does not adversely affect the consequences of any accident previously evaluated.

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

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

   Response: No.

   The proposed change revises the definition of SDM. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The change does not alter assumptions made in the safety analysis regarding SDM.

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

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

Response: No.

The proposed change revises the definition of SDM. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change ensures that the SDM assumed in determining safety limits, limiting safety system settings or limiting conditions for operation is correct for all BWR fuel types at all times during the fuel cycle.

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. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Acting Branch Chief: R. Beall

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

Date of amendment request: April 24, 2012, as supplemented by letters dated July 12 and August 23, 2012, and January 14, February 12, March 13, and June 13, 2013.

Description of amendment request: The proposed amendment would adopt National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants" (2001 Edition). Implementation of the regulatory actions presented in the attachments to the license amendment request will enable Cooper Nuclear Station to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a), 10 CFR 50.48(c), and the guidance in Regulatory Guide (RG) 1.205, Revision 1.

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

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

   Response: No.

   Operation of the Cooper Nuclear Station (CNS) in accordance with the proposed amendment does not result in a significant increase in the probability or consequences of accidents previously evaluated. The proposed amendment does not affect accident initiators or precursors as described in the CNS Updated Safety Analysis Report (USAR), nor does it adversely alter design assumptions, conditions, or configurations of the facility, and it does not adversely impact the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the way in which safety-related systems perform their functions as required by the accident analysis. The SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition will remain capable of performing their design functions.

   The purpose of this amendment is to permit CNS to adopt a new risk-informed, performance-based fire protection licensing basis that complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c), as well as the guidance contained in Regulatory Guide (RG) 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection requirements that are an acceptable alternative to the 10 CFR part 50, appendix R, fire protection features (69 FR 33536; June 16, 2004). Engineering analyses, which may include engineering evaluations, probabilistic risk assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based requirements of NFPA 805 have been met.

NFPA 805, taken as a whole, provides an acceptable alternative for satisfying General Design Criterion 3 (GDC 3) of appendix A to 10 CFR part 50. It meets the underlying intent of the NRC's existing fire protection regulations and guidance, and achieves defense-in-depth along with the goals, performance objectives, and performance criteria specified in NFPA 805, Chapter 1. In addition, if there are any increases in core damage frequency (CDF) or risk as a result of the transition to NFPA 805, the increase will be small, governed by the delta risk requirements of NFPA 805, and consistent with the intent of the Commission's Safety Goal Policy.

Based on the above, the implementation of this amendment to transition the Fire Protection Plan (FPP) at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), does not result in a significant increase in the probability of any accident previously evaluated. In addition, all equipment required to mitigate an accident remains capable of performing the assumed function.

Therefore, the consequences of any accident previously evaluated are not significantly increased with the implementation of this License Amendment Request.

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

Response: No.

Operation of CNS in accordance with the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. Any scenario or previously analyzed accident with offsite dose consequences was included in the evaluation of design basis accidents (DBA) documented in the USAR as a part of the transition to NFPA 805. The proposed amendment does not impact these accident analyses. The proposed change does not alter the requirements or functions for systems required during accident conditions, nor does it alter the required mitigation capability of the fire protection program, or its functioning during accident conditions as assumed in the licensing basis analyses and/or DBA radiological consequences evaluations.

The proposed amendment does not adversely affect accident initiators nor alter design assumptions, or conditions of the facility. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to maintain the unit in a safe and stable condition remain capable of performing their design functions.

The purpose of the proposed amendment is to permit CNS to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c) and the guidance in Revision 1 of RG 1.205. As indicated in the Statements of Consideration, the NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to identify fire protection systems and features that are an acceptable alternative to the 10 CFR part 50, appendix R fire protection features.

The requirements in NFPA 805 address only fire protection and the impacts of fire effects on the plant have been evaluated. The proposed fire protection program changes do not involve new failure mechanisms or malfunctions that could initiate a new or different kind of accident beyond those already analyzed in the USAR. Based on this, as well as the discussion above, the implementation of this amendment to transition the FPP at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), 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.

Operation of CNS in accordance with the proposed license amendment does not involve a significant reduction in a margin of safety. The transition to a new risk-informed, performance-based fire protection licensing basis that complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c) does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed license amendment does not adversely affect existing plant safety margins or the reliability of equipment assumed in the USAR to mitigate accidents. The proposed change does not adversely impact systems that respond to safely shut down the plant and maintain the plant in a safe shutdown condition. In addition, the proposed license amendment will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without implementation of appropriate compensatory measures.

The purpose of the proposed license amendment is to permit CNS to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and 10 CFR 50.48(c) and the guidance in Regulatory Guide 1.205. The NRC considers that NFPA 805 provides an acceptable methodology and performance criteria for licensees to

> identify fire protection systems and features that are an acceptable alternative to the 10 CFR part 50, appendix R required fire protection features (69 FR 33536; June 16, 2004).
>
> The risk evaluations for plant changes, in part as they relate to the potential for reducing a safety margin, were measured quantitatively for acceptability using the delta risk guidance contained in RG 1.205. Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling calculations, have been performed to demonstrate that the performance-based methods of NFPA 805 do not result in a significant reduction in the margin of safety.
>
> As such, the proposed changes are evaluated to ensure that risk and safety margins are kept within acceptable limits. Based on the above, the implementation of this amendment to transition the FPP at CNS to one based on NFPA 805, in accordance with 10 CFR 50.48(c), will not significantly reduce 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.

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: February 28, 2013.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3.6.5, "Containment Air Temperature," to increase the allowable containment average temperature from 120°F to 125°F. The revised TS Section 3.6.5 would read as follows: "Containment average air temperature shall be ≤ 125 °F."

The licensee supports the proposed change by revising the analyses for Loss of Coolant Accident (LOCA) and a Main Steam Line break, and evaluating the containment response by either increase in initial containment air temperature or increase in the temperature of safety injection accumulators, which are located in the Ginna containment, and are expected to be at the same temperature as containment air.

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 to increase the containment average air temperature limit to 125 °F, from 120 °F, does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change. This proposed change will not affect radiological dose consequence analyses. The radiological dose consequence analyses assume a certain containment atmosphere leak rate based on the maximum allowable containment leakage rate, which is not affected by the change in allowable average containment air temperature resulting in a higher calculated peak containment pressure. The 10 CFR part 50, appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The acceptable leakage corresponds to the peak allowable containment pressure of 60 psig. The radiological dose consequence analyses assume a certain source term, which is not affected by the change in allowable average containment air temperature. All core limitations set forth in 10 CFR 50.46 continue to be met. The consequences of an accident previously evaluated will not be increased by this proposed change.

   Therefore, operation of the facility in accordance with the proposed change to the containment average air temperature limit will 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 for a higher allowable containment average air temperature to that currently in the TS Section 3.6.5. The calculated peak containment temperature and pressure remain below the containment design temperature and pressure of 286 °F and 60 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, operation of the facility in accordance with the proposed change to TS Section 3.6.5 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 calculated peak containment pressure and temperature remain below the containment design pressure and temperature of 60 psig and 286 °F, respectively. The penalties applied to the BE LBLOCA [best estimate loss of coolant accident] analysis result in the limitations set forth in 10 CFR 50.46 continuing to be met. Since the radiological consequence analyses are based on the maximum allowable containment leakage rate, which is not being revised, the change in the calculated peak containment pressure and temperature and changes in core response do not represent a significant change in the margin of safety. The longterm impact of the peak containment temperature following a design basis accident exceeding the EQ profile by 2 °F with respect to the current licensing basis is negligible.

Therefore, operation of the facility in accordance with the proposed change to increase the allowable containment average air temperature from 120 °F to 125 °F does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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: Robert Beall

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

Date of amendment request: October 3, 2013.

Description of amendment request: The proposed amendment would revise the scheduled completion date of the Cyber Security Plan Milestone 8.

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 revises the Cyber Security Plan Implementation Schedule. This change does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected.

   The proposed change is a change to the completion date of implementation milestone 8 that in itself does not require any plant modifications which affect the performance capability of the structures, systems, and components relied upon to mitigate the consequences of postulated accidents and have no impact on the probability or consequences of an accident previously evaluated.

   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 revises the Cyber Security Plan Implementation Schedule. This proposed change to modify the completion date of implementation milestone 8 does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected. The proposed change does not require any plant modifications which affect the performance capability of the structures, systems and components relied upon to mitigate the consequences of postulated accidents. This change also does not create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications. The proposed change revises the Cyber Security Plan Implementation Schedule. Because there is no change to these established safety margins as result of this change, the proposed change does not involve a significant reduction in a margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

<u>Attorney for licensee</u>: J. Hagood Hamilton, Jr., South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

<u>NRC Branch Chief</u>: Robert J. Pascarelli.

<u>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</u>

Date of amendment request: October 2, 2013.

Description of amendment request: The proposed change would amend Combined License Nos. NPF-93 and NPF-94 for the Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 by departing from the plant-specific Design Control Document (DCD) Tier 1 (and corresponding Combined License Appendix C information) and Tier 2 material by making changes to the Non-Class 1E dc and Uninterruptible Power Supply System (EDS) and Uninterruptible Power Supply System (IDS) and making changes to the corresponding Tier 1 information in Appendix C to the Combined License. The proposed changes would:

(1) Increase EDS total equipment capacity, component ratings, and protective device sizing to support increased load demand,

(2) Relocate equipment and moving Turbine Building (TB) first bay EDS Battery Room and Charger Room. The floor elevation increases from elevation 148’-0” to elevation 148’-10” to accommodate associated equipment cabling with this activity, and

(3) Remove the Class 1E IDS Battery Back-up tie to the Non-Class 1E EDS Battery.

Because, this proposed change requires a departure from Tier 1 information in the Westinghouse Advanced Passive 1000 DCD, the licensee also requested an exemption from the requirements of the Generic DCD Tier 1 in accordance with 10 CFR 52.63(b)(1).

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

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

   Response: No.

   The design function of the Turbine Building (TB) is to provide weather protection for the laydown and maintenance of major turbine/generator components. The TB first bay is a seismic Category II structure designed to prevent the collapse under a safe shutdown earthquake (SSE) to protect the adjacent auxiliary building. The electrical system and air-handling units are designed to provide electrical power to plant loads and maintain acceptable temperatures for electrical equipment rooms and work areas. The electrical equipment continues to be in accordance with the same codes and standards stated in the Updated Final Safety Analysis Report (UFSAR). The proposed relocation of equipment, including the increase in floor elevation by 10 inches to accommodate overhead equipment cabling, does not impact the TB design function. The TB first bay continues to meet seismic Category II requirements. Based on this, the proposed changes would not increase the probability of an accident previously evaluated.

The proposed changes do not involve any accident initiating event, thus the probabilities of the accidents previously evaluated are not affected. The relocation of equipment does not involve any safety-related structures, systems, or components; the affected rooms do not represent a radioactive material barrier; and this activity does not affect the containment of radioactive material. The radioactive material source terms and release paths used in the safety analyses are unchanged, thus the radiological releases in the accident analyses are not affected. Therefore, the consequences of an accident previously evaluated are not affected.

Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes would use the same type of electrical equipment with higher ratings and capacity, change the source of a battery back-up, and relocate equipment. The electrical equipment will continue to perform its design functions because the same electrical codes and standards as stated in the UFSAR continue to be met. Therefore the proposed changes do not affect equipment failure probabilities or alter any accident initiator or initiating sequence of events. The proposed changes in location of equipment and elevation of the TB first bay floor do not affect the design function of the TB first bay to protect the adjacent auxiliary building by meeting seismic Category II structure requirements, or affect the operation of the relocated equipment, or the ability of the relocated equipment to meet its design functions. Because the SSCs and equipment affected by the proposed changes continue to meet their design functions, the structural codes and standards as stated in the UFSAR, the proposed changes do not introduce a different type of accident than those previously considered.

   Therefore, this activity 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 current seismic requirements applicable to the seismic Category II TB first bay structure, including the seismic modeling and analysis methods, will continue to apply to the TB first bay floor elevation increase. The proposed changes to relocate equipment and the increase in the floor elevation will continue to meet the fire rating requirements and will be in accordance with the same codes and standards currently identified in the UFSAR. The proposed changes to the electrical equipment will continue to meet existing electrical equipment industry standard recommendations identified in the UFSAR. Because no safety analysis or design basis acceptance limit/criterion is challenged or exceeded by these proposed changes, no margin of safety is reduced.
>
> Therefore, the changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Lawrence Burkhart.

Dated at Rockville, Maryland, this 15th day of November 2013.

For the Nuclear Regulatory Commission.

Michele G. Evans, Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation

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