NUCLEAR REGULATORY COMMISSION

[NRC–2012–0131]

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 (the Commission or 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 May 17, 2012 to May 30, 2012. The last biweekly notice was published on May 29, 2012 (77 FR 31655).

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–2012–0131. You may submit comments by the following methods:

- *Federal Rulemaking Web site:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2012–0131. Address questions about NRC dockets to Carol Gallagher; telephone: 301–492–3668; email: *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–2012–0131 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–2012–0131.
- *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 email 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–2012–0131 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 Title 10 of the *Code of Federal Regulations* (10 CFR) 50.92, 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 ''Rules of Practice for Domestic Licensing Proceedings'' 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 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 email 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 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 NRC guidance available on the NRC 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 email notice confirming receipt of the document. The

E-Filing system also distributes an email notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the 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 Web site at *http://www.nrc.gov/site-help/e-submittals.html*, by email 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. 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. Non-timely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

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 email to *pdr.resource@nrc.gov*.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendment request: March 8, 2012.

Description of amendment request: The amendments would eliminate the use of the term CORE ALTERATIONS throughout the Technical Specifications (TSs). The proposed amendment incorporates changes reflected in Technical Specification Task Force (TSTF) Change Traveler TSTF–471–A, Revision 1, ''Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes.'' The U.S. Nuclear Regulatory Commission (NRC) staff reviewed and approved TSTF–471 by letter dated December 7, 2006 (ADAMS Accession No. ML062860320). The changes are consistent with NUREG–1432, ''Standard Technical Specifications—Combustion Engineering Plants,'' Revision 4 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12102A165).

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

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

Response: No.

The proposed change eliminates the use of the defined term CORE ALTERATIONS from the Technical Specifications. CORE ALTERATIONS are not an initiator of any accident previously evaluated except a fuel handling accident. The revised Technical Specifications that protect the initial conditions of a fuel handling accident also require the suspension of movement of irradiated fuel assemblies. Suspending movement of irradiated fuel assemblies protects the initial condition of a fuel handling accident and, therefore, suspension of CORE ALTERATIONS is not required. Suspension of CORE ALTERATIONS does not provide mitigation of any accident previously evaluated. Therefore, CORE ALTERATIONS do not affect the initiators of the accidents previously evaluated and suspension of CORE ALTERATIONS does not affect the mitigation of the accidents 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.

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

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

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

Response: No.

Only two accidents are postulated to occur during plant conditions where CORE ALTERATIONS may be made: a fuel handling accident and a boron dilution accident. Suspending movement of irradiated fuel assemblies prevents a fuel handling accident. Also requiring the suspension of CORE ALTERATIONS is a redundant requirement to suspending movement of irradiated fuel assemblies and does not increase the margin of safety. CORE ALTERATIONS have no effect on a boron dilution accident. Core components are not involved in the initiation or mitigation of a boron dilution accident and the SHUTDOWN MARGIN limit is based on assuming the worse-case configuration of the core components.

Therefore, CORE ALTERATIONS have no effect on the margin of safety related to a boron dilution accident.

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

Attorney for licensee: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072–2034.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: April 13, 2012.

Description of amendment request: The proposed amendment would revise the Millstone Power Station, Unit 2 (MPS2) Technical Specification (TS) requirements related to diesel fuel oil testing consistent with NUREG–1432, Rev. 3.1, ''Standard Technical Specifications, Combustion Engineering Plants,'' December 1, 1995, and NRC approved Technical Specification Task Force (TSTF) TSTF–374, ''Revision to TS 5.5.13 and Associated TS Bases for Diesel Fuel Oil,'' Revision 0.

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

Criterion 1

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

Response: No.

The proposed changes modify the TS requirements related to diesel fuel oil testing consistent with NRC approved TSTF–374, ''Revision to TS 5.5.13 and Associated TS Bases for Diesel Fuel Oil,'' Revision 0. To adopt changes consistent with the content of TSTF–374 for use in the custom TS of MPS2, the existing MPS2 diesel fuel oil testing program will be modified. These changes replace the criteria of ''Water and sediment < 0.05%'' with the criteria of ''A clear and bright appearance with proper color or a water and sediment content within limits'' and remove specific American Society for Testing and Materials (ASTM) standard references from TS.

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

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

Criterion 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 are used to provide operational flexibility regarding evolving industry standards while maintaining operational conditions which are consistent with the design basis. Removing of specific details from TS, since the details are already specified in licensee-controlled documents, provides the flexibility needed to maintain state-of-the-art technology in fuel oil sampling and analysis methodology. The procedural details associated with the involved specifications that are removed from TS and residing in licensee-controlled documents are not required to be in the TS to provide adequate protection of the public health and safety, since the TS still retains the requirement for compliance with applicable standards. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation in the provision, maintaining, or use of diesel fuel oil. The requirements retained in the TS continue to require testing of the diesel fuel oil to ensure the proper functioning of the DGs.

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

Criterion 3

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

Response: No.

The proposed changes are consistent with the content of TSTF–374 for use in the custom TS of MPS2. These changes remove specific ASTM standard references and a preventive maintenance cleaning requirement from TS since the references and requirements are already specified in licensee-controlled documents. The proposed changes provide the flexibility needed to improve fuel oil sampling and analysis methodologies while maintaining sufficient controls to ensure continued quality of the fuel oil. The margin of safety provided to the DGs by these detailed fuel specifications is unaffected by the proposed changes since there continue to be TS requirements to ensure fuel oil is of the appropriate quality for emergency DG use and DG operability is unaffected.

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: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS–2, Richmond, VA 23219.

NRC Branch Chief: George Wilson.

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

Date of amendment request: March 20, 2012.

Description of amendment request: The proposed amendment would modify Braidwood and Byron Technical Specifications to permanently exclude portions of the steam generator (SG) tube below the top of the SG tubesheet from periodic SG tube inspections and plugging or repair for Braidwood, Unit 2 and for Byron, Unit 2. In addition, the proposed amendment would revise TS 5.6.9 to remove reference to the previous temporary alternate repair criteria and provide reporting requirements specific to the permanent alternate repair criteria.

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 previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator (SG) inspection and reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Of the various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR), postulated steam line break (SLB), feedwater line break (FLB), locked rotor and control rod ejection accident evaluations. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model D5 SGs has shown that axial loading of the tubes is negligible during an SSE.

During the SGTR event, the required structural integrity margins of the SG tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the presence of the tubesheet and the tube-to-tubesheet joint. Tube burst cannot occur within the thickness of the tubesheet. The tube-to-tubesheet joint constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side, and tubesheet rotation. Based on this design, the structural margins against burst, as discussed in draft Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' and TS 5.5.9, are maintained for both normal and postulated accident conditions.

The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural and leakage integrity of the SG tubes consistent with the performance criteria of TS 5.5.9. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from tube degradation below the proposed limited inspection depth is limited by the tube-to-tubesheet crevice. Consequently, negligible normal operating leakage is expected from degradation below the inspected depth within the tubesheet region.

The consequences of an SGTR event are not affected by the primary-to-secondary leakage flow during the event as primary-to-secondary leakage flow through a postulated tube that has been pulled out of the tubesheet is essentially equivalent to a severed tube. Therefore, the proposed change does not result in a significant increase in the consequences of a SGTR.

Primary-to-secondary leakage from tube degradation in the tubesheet area during operating and accident conditions is restricted due to contact of the tube with the tubesheet. The leakage is modeled as flow through a porous medium through the use of the Darcy equation. The leakage model is used to develop a relationship between operational leakage and leakage at accident conditions that is based on differential pressure across the tubesheet and the viscosity of the fluid. A leak rate ratio was developed to relate the leakage at operating conditions to leakage at accident conditions. Since the fluid viscosity is based on fluid temperature and it is shown that for the most limiting accident, the fluid temperature does not exceed the normal operating temperature and therefore the viscosity ratio is assumed to be 1.0. Therefore, the leak rate ratio is a function of the ratio of the accident differential pressure and the normal operating differential pressure.

The leakage factor of 1.93 for Braidwood Station Unit 2 and Byron Station Unit 2, for a postulated SLB/FLB, has been calculated as shown in Table 9–7 of WCAP–17072–P, Revision 0. However, EGC Braidwood Station Unit 2 and Byron Station Unit 2 will apply a factor of 3.11 as determined by Westinghouse evaluation LTR–SGMP–09–100 P-Attachment, Revision 1, to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). The leakage factor of 3.11 applies specifically to Byron Unit 2 and Braidwood Unit 2, both hot and cold legs, in Table RAI24–2 of LTR–SGMP–09–100 P-Attachment, Revision 1. Through application of the limited tubesheet inspection scope, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. The assumed accident induced leak rate limit is 0.5 gallons per minute at room temperature (gpmRT) for the faulted SG and 0.218 gpmRT for each of the unfaulted SGs for accidents that assume a faulted SG. These accidents are the SLB and the locked rotor with a stuck open PORV. The assumed accident induced leak rate limit for accidents that do not assume a faulted SG is 1.0 gpmRT for all SGs. These accidents are the locked rotor and control rod ejection.

No leakage factor will be applied to the locked rotor or control rod ejection transients due to their short duration, since the calculated leak rate ratio is less than 1.0.

The TS 3.4.13 operational leak rate limit is 150 gallons per day (gpd) (0.104 gpmRT) through any one SG. Consequently, there is sufficient margin between accident leakage and allowable operational leakage. The maximum accident leak rate ratio for the Model D5 design SGs is 1.93 as indicated in WCAP–17072–P, Revision 0, Table 9–7. However, EGC will use the more conservative value of 3.11 accident leak rate ratio for the most limiting SG model design identified in Table RAI24–2 of LTRSGMP–09–100 P-Attachment Revision 1. This results in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (0.5 gpmRT).

For the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.11 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the OA, the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.11 and compared to the observed operational leakage.

Based on the above, the performance criteria of NEI–97–06, Revision 3, and draft RG 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of the applicable accidents 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 introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the permanent alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

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

Response: No.

The proposed change defines the safety significant portion of the SG tube that must be inspected and repaired. WCAP–17072–P, Revision 0, as modified by WCAP–17330–P, Revision 1, identifies the specific inspection depth below which any type tube degradation has no impact on the performance criteria in NEI 97–06, Revision 3, 'Steam Generator Program Guidelines.''

The proposed change that alters the SG inspection and reporting criteria maintains the required structural margins of the SG tubes for both normal and accident conditions. NEI 97–06, and draft RG 1.121 are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. Draft RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, ''Reactor Coolant Pressure Boundary,'' GDC 15, ''Reactor Coolant System Design,'' GDC 31, ''Fracture Prevention of Reactor Coolant Pressure Boundary,'' and GDC 32, ''Inspection of Reactor Coolant Pressure Boundary,'' by reducing the probability and consequences of a SGTR. Draft RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SGTR are reduced. This draft RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, WCAP–17072–P, Revision 0, as modified by WCAP–17330–P, Revision 1, defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage as described in WCAP–17072–P, Revision 0, as modified by LTR–SGMP–09–100 P–Attachment\ shows that significant margin exists between an acceptable level of leakage during normal operating conditions that ensures meeting the SLB accident-induced leakage assumption and the TS leakage limit of 150 gpd.

Based on the above, it is concluded that the proposed changes do not result in any reduction in a margin of safety.

Based on the above, EGC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(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 Branch Chief: Jacob I. Zimmerman.

Luminant Generation Company LLC, Docket Nos. 50–445 and 50–446, Comanche Peak Nuclear Power Plant, Units 1 and 2, Somervell County, Texas

Date of amendment request: March 28, 2012.

Brief description of amendment: The amendment would revise Technical Specification (TS) 5.5.9, ''Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program,'' to permanently exclude portions of the Comanche Peak Nuclear Power Plant (CPNPP), Unit 2, Model D5 SG tubes below the top of the SG tubesheet from periodic SG tube inspections. In addition, this amendment would revise TS 5.6.9, ''Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report,'' to provide permanent reporting requirements specific to CPNPP, Unit 2, that have previously been established on a one-cycle basis.

The proposed amendment constitutes a redefinition of the SG tube primary-to-secondary pressure boundary and defines the safety significant portion of the tube that must be inspected or plugged. Tube flaws detected below the safety significant portion of the tube are not required to be plugged. Allowing flaws in the non-safety significant portion of the tube to remain in service minimizes unnecessary tube plugging and maintains the safety margin of the steam generators to perform the safety function to maintain the reactor coolant pressure boundary, maintain reactor coolant flow, and maintain primary to secondary heat transfer.

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.

Of the accidents previously evaluated, the limiting transients with consideration to the proposed change to the SG tube inspection and repair criteria are the steam generator tube rupture (SGTR) event, the steam line break (SLB), and the feed line break (FLB) postulated accidents.

The required structural integrity margins of the SG tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the presence of the tubesheet and the tube-to-tubesheet joint. Tube burst cannot occur within the thickness of the tubesheet. The tube-to-tubesheet joint constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, differential pressure between the primary and secondary side, and tubesheet rotation. Based on this design, the structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR [Pressurized Water Reactor] Steam Generator Tubes,'' [(Agencywide Documents Access and Management System (ADAMS) Accession No. ML082120667)] and TS 5.5.9 are maintained for both normal and postulated accident conditions.

The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural and leakage integrity of the SG tubes consistent with the performance criteria in TS 5.5.9. Therefore, the proposed change results in no significant increase in the probability of the occurrence of [an] SGTR accident.

At normal operating pressures, leakage from tube degradation below the proposed limited inspection depth is limited by the tube-to-tubesheet crevice. Consequently, negligible normal operating leakage is expected from degradation below the inspected depth within the tubesheet region. The consequences of an SGTR event are not affected by the primary-to-secondary leakage flow during the event as primary-to-secondary leakage flow through a postulated tube that has been pulled out of the tubesheet is essentially equivalent to a severed tube. Therefore, the proposed change does not result in a significant increase in the consequences of [an] SGTR.

The probability of [an] SLB is unaffected by the potential failure of a steam generator tube as the failure of tube is not an initiator for [an] SLB event.

The leakage factor of 3.16 for CPNPP Unit 2, for a postulated SLB/FLB, has been calculated as described in Westinghouse [Electric Company, LLC] Letter LTR–SGMP–09–100 [N]P—Attachment, ''Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators,'' dated August 12, 2009 [(ADAMS Accession No. ML101730391)], and is shown in Revised Table 9–7 of this same document. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.16 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.16 and compared to the observed operational leakage. The accident-induced leak rate limit for CPNPP Unit 2 is 1.0 gpm [gallons per minute]. The TS operational leak rate limit through any one steam generator is 150 gpd [gallons per day] (0.1 gpm). Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB overall leakage factor is 3.16 resulting in significant margin between the conservatively estimated accident induced leakage and the allowable accident leakage.

No leakage factor was applied to the locked rotor or control rod ejection transients due to their short duration.

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the SG inspection and reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

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

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

Response: No.

The proposed change that alters the steam generator inspection and reporting criteria does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses.

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 the margin of safety?

Response: No.

The proposed change that alters the steam generator inspection and reporting criteria maintains the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute [(NEI) document NEI] 97–06, Rev. 3, ''Steam Generator Program Guidelines,'' and NRC Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, ''Reactor coolant pressure boundary,'' GDC 15, ''Reactor coolant system design,'' GDC 31, ''Fracture prevention of reactor coolant pressure boundary,'' and GDC 32, ''Inspection of reactor coolant pressure boundary,'' by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SGTR are reduced. RG 1.121 uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, the H* Analysis documented in Section 4.1 [of the application dated March 28, 2012] defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

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

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

Attorney for licensee: Timothy P. Matthews, Esq., Morgan, Lewis and Bockius, 1800 M Street NW., Washington, DC 20036.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: September 29, 2011, as supplemented by letter dated March 12, 2012.

Description of amendment request: The proposed amendment would revise the DAEC Technical Specifications (TSs) by modifying existing Surveillance Requirements (SRs) regarding various modes of operation of the main steam safety/relief valves (SRVs). The proposed amendment would modify the TS requirements for testing of the SRVs by replacing the current requirement to manually actuate each SRV during plant startup with a series of overlapping tests that demonstrate the required functions of successive valve stages. Elimination of the manual actuation requirement at low reactor pressure and steam flow decreases the potential for SRV leakage and spurious SRV opening.

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

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

Response: No.

The proposed changes modify TS SR 3.4.3.2, SR 3.5.1.9, and SR 3.6.1.5.1 to provide an alternative means for testing the main steam SRVs, ADS [Automatic Depressurization System] valves, and LLS [Low-Low Set] relief valves. Accidents are initiated by the malfunction of plant equipment, or the catastrophic failure of plant structures, systems, or components. The performance of SRV testing is not a precursor to any accident previously evaluated and does not change the manner in which the valves are operated. The proposed testing requirements will not contribute to the failure of the SRVs nor any plant structure, system, or component. NextEra Energy Duane Arnold has determined that the proposed change in testing methodology provides an equivalent level of assurance that the SRVs are capable of performing their intended safety functions. Thus, the proposed changes do not affect the probability of an accident previously evaluated.

The performance of SRV testing provides confidence that the relief valves are capable of depressurizing the reactor pressure vessel (RPV). This will protect the reactor vessel from overpressurization and allow the combination of the Low Pressure Coolant Injection and Core Spray Systems to inject into the RPV as designed. The LLS relief logic causes two LLS relief valves to be opened at a lower pressure than the relief mode pressure setpoints and causes the LLS relief valves to stay open longer, such that reopening of more than one valve is prevented on subsequent actuations. Thus, the LLS relief function prevents excessive short duration SRV cycling, which limits induced thrust loads on the SRV discharge line for subsequent actuations of the relief valve. The proposed changes do not affect any function related to the safety mode of the dual function SRVs. The proposed changes involve the manner in which the subject valves are tested, and have no effect on the types or amounts of radiation released or the predicted offsite doses in the event of an accident. The proposed testing requirements are sufficient to provide confidence that these valves are capable of performing their intended safety functions.

In addition, an inadvertent opening of an SRV is an analyzed event in the DAEC UFSAR [Updated Final Safety Analysis Report] (Section 15.1.7.2), as well as the assumption of a single SRV failure to open on demand in other transients and accidents, as appropriate (e.g., one ADS valve failure in the LOCA [loss-of-coolant accident] analysis). Since the proposed testing requirements do not alter the assumptions for any analyzed transient or accident, the radiological consequences of any accident previously evaluated are not increased.

Therefore, the 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 previously evaluated?

Response: No.

The proposed changes do not affect the assumed accident performance of the main steam SRVs, nor any plant structure, system, or component previously evaluated. The proposed changes do not install any new equipment, and installed equipment is not being operated in a new or different manner. The proposed change in test methodology will ensure that the valves remain capable of performing their safety functions due to meeting the testing requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, with the exception of opening the valve following installation or maintenance for which a relief request has been submitted (Ref. 6.1 [of the September 29, 2011, application]), proposing an acceptable alternative. No setpoints are being changed which would alter the dynamic response of plant equipment. Accordingly, no new failure modes are introduced.

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

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

Response: No.

Overpressure protection of the RCPB [reactor coolant pressure boundary] is based on the SRVs' setpoints and total relief capacity. The setpoints are verified at an offsite testing facility; this requirement is not altered by the proposed change. The relief capacity of each SRV is determined by the valve's geometry, which is also not altered by the proposed test methods.

The proposed changes will allow testing of the valve actuation electrical circuitry, including the solenoid, and mechanical actuation components, without causing the SRV to open. The SRVs will be manually actuated prior to installation in the plant. Therefore, all modes of SRV operation will be tested prior to entering the mode of operation requiring the valves to perform their safety functions. The proposed changes do not affect the valve setpoint or the operational criteria that cause the SRVs to open during plant transients or accidents, either manually or automatically. There are no changes proposed which alter the setpoints at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation.

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. Mitchell S. Ross, P. O. Box 14000 Juno Beach, FL 33408–0420.

NRC Acting Branch Chief: Istvan Frankl.

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

Date of amendment request: April 6, 2012, and revised on April 12 and May 7, 2012.

Description of amendment request: The proposed changes would amend Combined License Nos. NPF–91 and NPF–92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively, in regard to the upper tolerance on the Nuclear Island (NI) critical sections basemat thickness as identified in the plant-specific Design Control Document (DCD).

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.

As indicated in FSAR (plant-specific DCD) Subsection 3.8.5.5, the design function of the basemat is to provide the interface between the nuclear island structures and the supporting soil or rock. The basemat transfers the load of nuclear island structures to the supporting soil or rock. The basemat transmits seismic motions from the supporting soil or rock to the nuclear island. The revision of the basemat construction tolerance does not have an adverse impact on the response of the basemat and nuclear island structures to safe shutdown earthquake ground motions or loads due to anticipated transients or postulated accident conditions. The revision of the basemat construction tolerance 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 normal operation or 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, 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 change is to increase the construction tolerance for the basemat thickness. The revision of the basemat construction tolerance does not change the design of the basemat or nuclear island structures. The revision of the basemat construction tolerance does not change the design function, support, design, or operation of mechanical and fluid systems. The revision of the basemat construction tolerance does not result in a new failure mechanism for the basemat or new accident precursors. As a result, the design function of the basemat is not adversely affected by the proposed change.

Therefore, the proposed change will 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 revision in the basemat thickness construction tolerance does not have an adverse impact on the strength of the basemat. The increase in the basemat thickness construction tolerance does not have an adverse impact on the seismic design spectra or the structural analysis of the basemat or other nuclear island structures. The revision in the basemat thickness construction tolerance has no impact of the analysis of the nuclear island for sliding or overturning. As a result, the design function of the basemat is not adversely affected by the proposed change.

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

NRC Branch Chief: Mark E. Tonacci.

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

Date of amendment request: April 2, 2012.

Description of amendment request: The proposed amendment would delete the Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch Reactor Trip Function from the Technical Specification (TS) Table 3.3.1–1 Item 15.

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:

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

The initiating conditions and assumptions for accidents described in the Updated Final Safety Analyses Report remain as previously analyzed. The proposed change does not introduce a new accident initiator nor does it introduce changes to any existing accident initiators or scenarios described in the Updated Final Safety Analyses Report. The Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip is not credited for accident mitigation in any accident analyses described in the Updated Final Safety Analyses Report. The Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip was designed to meet the control and protection systems interaction criteria of IEEE–279. The Steam Generator Level Median Signal Selector (MSS) prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip to satisfy the IEEE–279 requirements. As such, the affected control and protection systems will continue to perform their required functions without adverse interaction, and maintain the capability to shut down the reactor when required on Low-Low Steam Generator water level. The ability to mitigate a loss of heat sink accident previously evaluated is unaffected. The frequency categories of previously evaluated accidents are not changed.

Therefore, neither the probability of occurrence nor the consequences of an accident previously evaluated is significantly increased.

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

The substitution of the MSS for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip will not introduce any new failure modes to the required protection functions. The MSS only interacts with the feedwater control system. The Steam Generator Water Level Low-Low protection function is not affected by this change. Isolation devices upstream of the MSS circuitry ensure that the Steam Generator Water Level Low-Low protection function is not affected. The MSS is designed to reduce the frequency of system failures through utilization of highly reliable components in a configuration that relies on a minimum of additional equipment. Components used in the MSS are of a quality consistent with low failure rates and minimum maintenance requirements, and conform to protection system requirements. Furthermore, the design provides the capability for complete unit testing that provides unambiguous determination of credible system failures. It is through these features that the overall design of the MSS minimizes the occurrence of undetected failures that may exist between test intervals.

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

Criterion 3—Does this change involve a significant reduction in a margin of safety?

The proposed amendment does not involve revisions to any safety analysis limits or safety system settings that will adversely impact plant safety. The proposed amendment does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this amendment revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The ability of the Steam Generator Water Level Low-Low reactor trip function credited in the safety analysis to protect against a sudden loss of heat sink event is not affected by the proposed change: Since the Steam Generator Low-Low Level trip is credited alone as providing complete protection for the accident transients that result in low steam generator level, eliminating the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip will not change any safety analysis conclusion for any analyzed accident described in the Updated Final Safety Analyses Report.

The MSS prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip and satisfies the IEEE–279 requirements.

The proposed change improves the margin of safety since removal of the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip function decreases the potential for challenges to plant safety systems, decreases the plant surveillance/maintenance activity, and reduces plant complexity. These changes result in a reduction in the potential for unnecessary plant transients.

The Technical Specifications continue to assure that the applicable operating parameters and systems are maintained within the design requirements and safety analysis assumptions. Therefore, the elimination of this trip function will not result in a significant reduction in the margin of safety as defined in the Updated Final Safety Analyses Report or Technical Specifications.

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

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

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS–2, Richmond, VA 23219.

NRC Branch Chief: Nancy L. Salgado.

Virginia Electric and Power Company, Docket No. 50–339, North Anna Power Station, Unit 2, Louisa County, Virginia

Date of amendment request: May 11, 2012.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 3.1.7, ''Rod Position Indication'' to allow two demand position indicators in one or more banks to be inoperable for up to 4 hours. This change is proposed as a temporary change to the TS for the current operating cycle and is proposed as a footnote to the current TS Limiting Condition for Operation (LCO) Section 3.1.7, Condition D.

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:

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

The proposed change provides a new Condition for two demand position indicators inoperable in one or more banks. The inoperability of two demand position indicators in one or more banks does not directly affect any accident analysis or design basis limits or cause any limit not to be met, because the demand position indicator only provides the intended demand as determined by the rod control system. The actual position of the control rods is determined by use of the Rod Position Indications (RPIs) for each control rod, or the movable incore detector system when the RPIs are inoperable.

The inoperability of the demand position indicators does prevent the comparison of the RPIs to the demand position indication for verification of rod insertion and rod group alignment limits, which is conducted as a periodic surveillance to maintain the reactor within analyzed conditions. The use of a 4 hour Completion Time limit provides a restriction that limits the time that reactor operation can continue during this loss of the demand position indication. Since the loss of the demand position indication does not cause the rods to change position, hence the actual control rod positions are expected to remain within required limits. Placing the Rod Control System in a condition incapable of rod movement is a positive control to prevent rod stepping while maintenance is being performed.

The proposed change to allow two demand position indicators to be inoperable in one or more banks does not affect the automatic or manual shutdown capability of the reactor protection system and no accident analyses are impacted by the proposed change. The operability of the control rods is not affected by the inoperability of the demand position indicators.

Therefore, neither the probability of occurrence nor the consequences of an accident previously evaluated is significantly increased.

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

The proposed change provides new requirements for two demand position indicators inoperable in one or more banks. No new accident initiators are introduced by the proposed requirements because the allowed condition for inoperability of the demand position indicators does not cause any new failure modes to be created that can cause an accident. The proposed change does not affect the reactor protection system or the reactor control system. The control rods should remain within the required limits because the failure of the demand position indicators does not cause the rods to change position and the RPIs remain available in the affected banks to verify the position of the control rods. In addition, the Rod Control System is placed in a condition incapable of rod movement as a positive control to prevent rod stepping while maintenance is being performed. Hence, no new failure modes or accident sequences are created that would cause a new or different kind of accident from any accident previously evaluated.

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

Criterion 3—Does this change involve a significant reduction in a margin of safety?

The operability of the RPIs 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 a new Condition for two demand position indicators inoperable in one or more banks. The inoperability of two demand position indicators for one or more banks results in the reduced ability to periodically verify that RPIs are operable and within expected limits. The condition does prevent the comparison of the RPIs to the demand position indication for verification of rod insertion and rod group alignment limits, which is conducted as periodic surveillance to maintain the reactor within analyzed conditions. The loss of the demand position indication does not cause the rods to change position, hence the actual control rod positions are expected to remain within required limits. The use of a 4 hour Completion Time limit provides a restriction that limits the time that reactor operation can continue during this loss of the demand position indication. This ensures the condition is promptly corrected or the reactor shutdown in accordance with the applicable Technical Specifications action statements. Thus, the proposed change maintains the operation of the reactor within the applicable margins of safety because the inoperability will be corrected or the unit will be shutdown prior to any significant reduction in the ability to verify control rod position by the use analog RPIs.

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

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

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS–2, Richmond, VA 23219.

NRC Branch Chief: Nancy L. Salgado.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: November 30, 2011.

Description of amendment request: The proposed amendment would revise the Wolf Creek Generating Station Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' Surveillance Requirements related to Diesel Generator test loads, voltage, and frequency. The proposed changes will correct non-conservative Diesel Generator load values that are currently under administrative controls.

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

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

Response: No.

The diesel generators are required to be OPERABLE in the event of a design basis accident coincident with a loss of offsite power to mitigate the consequences of the accident. The diesel generators are not accident initiators and therefore these changes do not involve a significant increase in the probability of an accident previously evaluated.

The accident analyses assume that at least one engineered safety feature bus is provided with power either from the offsite circuits or the diesel generators. The Technical Specification change proposed in this license amendment request will continue to assure that the diesel generators have the capacity and capability to assume their maximum design basis accident loads. The proposed change does not significantly change how the plant would mitigate an accident previously evaluated.

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

Therefore, the proposed change does not represent a significant increase 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 Technical Specification change does not involve a change in the plant design, system operation, or the use of the diesel generators. The proposed change requires the diesel generators to be tested at increased loads which envelope the actual power demand requirements for the diesel generators during design basis conditions. These revised loads continue to demonstrate the capability and capacity of the diesel generators to perform their required functions. There are no new failure modes or mechanisms created due to testing the diesel generators at the proposed test loading. Testing of the emergency diesel generators at the proposed test loadings does not involve any modification in the operational limits or physical design of plant systems. There are no new accident precursors generated due to the proposed test loadings.

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

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

Response: No.

The proposed Technical Specification change will continue to demonstrate that the diesel generators meet the Technical Specification definition of OPERABILITY, that is, the proposed tests will demonstrate that the diesel generators will perform their safety function and the necessary diesel generator attendant instrumentation, controls, cooling, lubrication and other auxiliary equipment required for the emergency diesel generators to perform their safety function loads are also tested at these proposed loadings. The proposed testing will also continue to demonstrate the capability and capacity of the diesel generators to supply their required loads for mitigating a design basis accident.

The proposed change 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 impacted by this change. The proposed change will not result in plant operation in a configuration outside the design basis.

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

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

Attorney for licensee: Jay Silberg, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street NW., Washington, DC 20037.

NRC Branch Chief: Michael T. Markley.

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 available online 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.*

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: July 27, 2011, as supplemented by letters dated September 16, 2011, and February 7, February 24, and April 3, 2012.

Brief description of amendment: The amendment modified River Bend Station's (RBS) Technical Specification (TS) 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' to revise the allowable value (AV) and related setpoints for the Main Steam Tunnel Temperature functions 1.e, 3.f, and 4.h in TS Table 3.3.6.1–1. In addition, the RBS's Emergency Action Levels will be revised to reflect the changes to the AV and related setpoints in TS 3.3.6.1.

Date of issuance: May 30, 2012.

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

Amendment No.: 174.

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

Date of initial notice in Federal Register: February 7, 2012 (77 FR 6147). The supplemental letters dated September 16, 2011, and February 7, February 24, and April 3, 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 May 30, 2012.

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: June 2, 2011, as supplemented by letter dated November 10, 2011.

Brief description of amendments: The amendments modify Technical Specification (TS) 3.1.2, ''Reactivity Anomalies,'' to change the method used to perform the reactivity anomaly surveillance. Specifically, the amendments allow performance of the surveillance based on the difference between the monitored (i.e., actual) core reactivity and the predicted core reactivity. The surveillance was previously performed based on the difference between the monitored control rod density and the predicted control rod density.

Date of issuance: May 25, 2012.

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

Amendments Nos.: 284 and 287.

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

Date of initial notice in ***Federal Register:*** September 6, 2011 (76 FR 55129).

The letter dated November 10, 2011, 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 May 25, 2012.

No significant hazards consideration comments received: No.

South Carolina Electric and Gas Company, Docket No. 50–395, Virgil C. Summer, Nuclear Station (VCSNS), Unit 1, Jenkinsville, South Carolina

Date of application for amendment: August 11, 2011.

Brief description of amendment: This amendment revised the VCSNS Technical Specification (TS) to allow an updating of the applicable topical report in TS 6.9.1.11, ''Core Operating Limits Report'' to use the three-dimensional Advanced Nodal Code neutronic model.

Date of Issuance: May 30, 2012.

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

Amendment No: 190.

Renewed Facility Operating License No. NPF–12: Amendment revises the License and Technical Specifications.

Date of initial notice in ***Federal Register:*** October 11, 2011 (76 FR 62864).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 2012.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 1st day of June, 2012.

For the Nuclear Regulatory Commission.

Michele G. Evans,

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

[FR Doc. 2012–13921 Filed 6–11–12; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket No. 50–443; NRC–2010–0206]

License Renewal Application for Seabrook Station, Unit 1 ; NextEra Energy Seabrook, LLC

AGENCY: Nuclear Regulatory Commission.

ACTION: License renewal application; intent to prepare supplement to draft