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[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2013-0060]

Biweekly Notice;

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

Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 7, 2013, to March 20, 2013. The last biweekly notice was published on March 19, 2013 (78 FR 16876).

ADDRESSES: You may access information and comment submissions related to this document, which the NRC possesses and is publicly available, by searching on http://www.regulations.gov under Docket ID **NRC-2013-0060**. You may submit comments by any of the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0060**. Address questions about NRC dockets to Carol Gallagher; telephone: 301-492-3668; e-mail: Carol.Gallagher@nrc.gov.

- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

- **Fax comments to:** RADB at 301-492-3446.

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

SUPPLEMENTARY INFORMATION:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID **NRC-2013-0060** when contacting the NRC about the availability of information regarding this document. You may access information related to this document, which the NRC possesses and is publicly available, by the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0060**.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly available documents online in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public

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

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

B. Submitting Comments

Please include Docket ID **NRC-2013-0060** 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 that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at *http://www.regulations.gov* as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

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

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

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

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

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

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

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

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

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

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

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

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

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

Information about applying for a digital ID certificate is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in the NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

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

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with the NRC guidance available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

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

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

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at http://ehd1.nrc.gov/ehd/, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. However, a request to intervene will require including information on local residence in order to demonstrate a proximity assertion of interest in the proceeding. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

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

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

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

Date of amendment request: July 21, 2010.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 3/4.9.3.1, "Decay Time" for Millstone Power Station Unit 2 (MPS2). The proposed change would revise TS 3/4.9.3.1 by reducing the minimum decay time for irradiated fuel prior to movement in the reactor vessel from 150 hours to 100 hours. A reduction in the minimum decay time requirement is requested to provide additional flexibility in outage planning such that irradiated fuel can be moved from the reactor vessel to the spent fuel pool earlier in an outage.

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
>
> Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
>
> Response: No.
>
> The accident of concern related to the proposed change is the FHA [fuel handling accident]. This accident assumes a dropped fuel assembly with resulting damage and release of the gap activity from the entire assembly. The FHA assumes that fuel movement is delayed for some time period after shutdown to accommodate for radioactive decay of the short-lived fission products. The probability of a FHA occurrence is dependent on moving fuel not when the fuel movement occurs. Reducing the decay time required by TS 3/4.9.3.1 from 150 hours to 100 hours does not increase the probability of a FHA since the timing of fuel movement in the reactor pressure vessel does not alter/impact the manner in which fuel assemblies are handled.
>
> Reducing the decay time requirement in TS 3/4.9.3.1 from 150 hours to 100 hours does not change the consequences of the offsite dose and control room dose projections for the currently approved design basis FHA analysis. The

current FHA analysis presented in FSAR [final safety analysis report] Section 14.7.4 and approved in License Amendment 298 assumes a minimum 100 hour decay time. Therefore, the dose results of this FHA analysis are unchanged, and remain within applicable regulatory limits.

Based on the reasons presented above, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

Will operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident will be created as a result of reducing the decay time requirement in TS 3/4.9.3.1. Plant operation, including fuel handling, will not be affected by the proposed change, as to when fuel is moved and no new failure modes will be created.

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

Criterion 3

Will operation of the facility in accordance with the proposed change involve a significant reduction in the margin of safety?

Response: No.

The proposed change does not significantly reduce the margin of safety. The current analysis of record for the FHA already accounts for irradiated fuel with at least 100 hours of decay. This approved analysis has shown that the projected doses will remain within applicable regulatory limits; therefore, the margin of safety is unchanged.

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 Acting Branch Chief: Sean C. Meighan.

Entergy Nuclear Operations, Inc., Docket No. 50-247, Indian Point Nuclear Generating, Unit 2, Westchester County, New York

Date of amendment request: January 28, 2013.

Description of amendment request: Nuclear Safety Advisory Letter 11-5 identified Westinghouse methodology errors in the long-term mass and energy releases during a large break loss-of-coolant accident. These impacted the containment integrity analysis for Indian Point, Unit 2. A re-analysis of the large break loss-of-coolant accident for the limiting single failure concluded that four, rather than three containment fan cooler units would need to be credited. The proposed change will revise Technical Specification Bases Sections 3.6.4, “Containment Pressure,” 3.6.5, “Containment Air Temperature,” and 3.6.6, “Containment Spray System and Containment Fan Cooler Unit (FCU) System.”

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 identified?

   Response: No.

   The proposed change would not change the current limiting EDG [emergency diesel generator] failure but would credit four rather than three can cooler units for containment heat removal. Four fan cooler units are available after the single failure. The fan cooler units are not accident initiators so the probability of an accident does not increase. Crediting all four fan cooler units will keep the post accident containment pressure within current limits and therefore does not increase the probability or consequences of a previously evaluated accident, but is a change from the analyses approved by the NRC [Nuclear Regulatory Commission] during stretch power uprate.

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

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

   Response: No.

   There are no changes to design, no changes to operating procedures, and the revised licensing basis change is consistent with the available equipment following the postulated worst case single failure.

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

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

   Response: No.

   The change reflects the credit for equipment that was always available but not previously credited (as a conservatism) in the licensing basis analyses. With credit for four fan cooler units, the post accident containment pressure remains within current limits and there is no 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.

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: Sean Meighan.

Entergy Nuclear Operations, Inc., Docket No. 50-247, Indian Point Nuclear Generating, Unit 2, Westchester County, New York

Date of amendment request: February 6, 2013.

Description of amendment request: The proposed amendment will revise the Reactor Heatup and Cooldown curves and Low Temperature Overpressure Protection Requirements in Technical Specifications (TSs) 3.4.3, "RCS [reactor coolant system] Pressure and Temperature (P/T) Limits," 3.4.6, "RCS Loops - MODE 4," 3.4.7, "RCS Loops - MODE 5, Loops Filled," 3.4.10, "Pressurizer Safety Valves," and 3.4.12, "Low Temperature Overpressure Protection (LTOP)."

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. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence of consequences of an accident previously evaluated.

   The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Except for a setpoint change for automatic PORV [power-operated relief valve] actuation, there are no physical changes to the plant being introduced by the proposed changes to the heatup and cooldown limitation curves. The proposed changes do not modify the RCS pressure boundary. That is, there are no changes in operating pressure, materials, or seismic loading. The proposed changes do not adversely affect the integrity of the RCS pressure boundary such that its function in the control of radiological consequences is affected. The proposed heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10CFR50 [10 CFR 50] Appendix G, and ASME B&PV code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code], Section XI, Appendix G edition with 2000 Addenda. The proposed heatup and cooldown limitation curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of RPV [reactor pressure vessel] beltline materials. Use of this methodology provides compliance with the intent of 10CFR50 [10 CFR 50] Appendix G and provides margins of safety that ensure non-ductile failure of the RPV will not occur. The proposed heatup and cooldown limitation curves prohibit operation in regions where it is possible for non-ductile failure of carbon and low alloy RCS materials to occur. Hence, the primary coolant pressure boundary integrity will be maintained throughout the limit of applicability of the curves, 48 EFPY [Effective Full Power Years].

Operation within the proposed LTOP limits ensures that overpressurization of the RCS at low temperatures will not result in component stresses in excess of those allowed by the ASME B&PV Code Section XI Appendix G.

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

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the heatup and cooldown limitation curves and the LTOP limits do not affect any activities or equipment other than the RCS pressure boundary and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

   The Proposed TS changes do not involve a significant reduction in the margin of safety. The revised heatup and cooldown limitation curves and LTOP limits are established in accordance with current regulations and the ASME B&PV Code 1998 edition with 2000 Addenda. These proposed changes are acceptable because the ASME B&PV Code maintains the margin of safety required by 10CFR50.55(a) [10 CFR 50.55(a)]. Because operation will be within these limits, the RCS materials will continue to behave in a non-brittle manner consistent with the original design bases.

> The proposed changes to the allowable operation of charging and safety injection pumps when LTOP is required to be operable is consistent with the IP2 licensing bases as established in TS Amendment 262.
>
> Therefore, Entergy has concluded that the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, and with the changes noted above in square brackets, 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: Sean Meighan.

Entergy Nuclear Operations, Inc., Docket No 50-286, Indian Point Nuclear Generating, Unit 3, Westchester County, New York

Date of amendment request: January 28, 2013.

Description of amendment request: Nuclear Safety Advisory Letter (NSAL) 11-5 identified Westinghouse methodology errors in the long-term mass and energy releases during a large break loss-of-coolant accident. These impacted the containment integrity analysis for Indian Point Unit No. 3 and required revisions to limiting initial operating conditions (i.e., containment temperature, containment pressure, and refueling water storage tank temperature) and require revisions to Technical Specifications (TSs) 3.5.4, "Refueling Water Storage Tank (RWST)," and 3.6.4, "Containment Pressure." In addition, revisions are proposed for TS 3.6.3, "Containment Isolation Valves," to delete a redundant surveillance requirement and TS 5.5.15, "Containment Leakage Rate Testing Program," to reflect a slightly higher calculated containment peak pressure.

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

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

   Response: No.

   The proposed change would not change the current EDG [emergency diesel generator] failure but limits the RWST temperature to ≤105°F and containment pressure to ≤1.5 psig [pounds per square inch gauge] (when RWST temperature is >95°F or containment/accumulator temperature is >125°F). The proposed change also removes a redundant TS for Containment testing and corrects the peak pressure in the containment testing program. The initial conditions assumed in accident analysis are not accident initiators so the probability of an accident does not increase. The change in initial conditions compensates for the error corrections and maintains the post accident containment pressure within 0.38 psig of the current value and within Containment testing limits and therefore does not increase the probability or consequences of a previously evaluated accident. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The change to the initial conditions assumed in the analysis for peak containment pressure, the removal of a redundant Technical Specification and the correction to the peak pressure limit in the Containment testing program do not create the possibility of a new or different accident. There are no changes to design or operating procedures that could create a new or different kind of accident since the changes only affect the initiating conditions. The revised analysis is consistent with the available equipment following the postulate worst case single failure.

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

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

   Response: No.

> The change in peak containment pressure is from 42 psig to 42.38 psig as a result of the error corrections of NSAL-11-5 and change to the initial conditions for the RWST temperature and containment pressure. There is an insignificant impact on other programs due to change in peak containment pressure, which remains well below the containment design pressure of 47 psig. Therefore there is not significant reduction in margin.
>
> Therefore the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Acting Branch Chief: Sean Meighan.

Exelon Generation Company (EGC), LLC, Docket No. 50-374, LaSalle County Station (LSCS), Unit 2, LaSalle County, Illinois

Date of amendment request: October 15, 2012.

Description of amendment request: The proposed amendment would remove License Conditions which are no longer necessary to address an interim configuration of the LaSalle County Station, Unit 2, spent fuel pool prior to completed installation of NETCO-SNAP-IN® inserts.

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

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

   Response: No.

   The proposed change removes License Conditions within the LSCS Unit 2 Operating License related to interim configurations of the SFP during the installation of the NETCO-SNAP-IN® inserts and the required completion date for installation. All changes proposed by EGC in this license amendment request are administrative in nature because they remove License Conditions that have either been satisfied or that are no longer applicable. There are no physical changes to the facilities, nor any changes to the station operating procedures, limiting conditions for operation, or limiting safety system settings.

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

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

   Response: No.

   The proposed change removes License Conditions within the LSCS Unit 2 Operating License related to interim configurations of the SFP during the installation of the NETCO-SNAP-IN® inserts and the required completion date for installation. There are no changes to the SFP criticality analysis associated with the proposed change. No physical changes to the plant are proposed, and there are no changes to the manner in which the plant is operated. Rather, the proposed change is administrative because it involves removing License Conditions that have either been satisfied or that are no longer applicable.

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

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

   Response: No.

   The proposed change removes License Conditions within the LSCS Unit 2 Operating License related to interim configurations of the SFP during the installation of the NETCO-SNAP-IN® inserts and the required completion date for installation. Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in Technical Specifications. The proposed change does not alter these established safety margins. The proposed

> change does not alter the criticality analysis for the SFP and does not affect the SFP criticality safety margin. The proposed change is administrative because it involves removing License Conditions that have either been satisfied or that are no longer applicable.
>
> 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 requested amendments involve no significant hazards consideration.

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

NRC Acting Branch Chief: Jeremy S. Bowen.

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: July 12, 2012.

Description of amendment request: The proposed amendments would modify Technical Specification 3.7.3, "Ultimate Heat Sink," by establishing controls which allow for the increase of cooling water temperature from 104°F to 107°F for plant safety systems.

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 makes no physical changes to the plant, nor does it alter any of the assumptions or conditions upon which the UHS is designed. These assumptions and conditions as described in the LSCS UFSAR include failure of the cooling lake dike, a loss of offsite power, and a DBA LOCA on one unit and a normal shutdown of the other unit.

The accidents analyzed in the UFSAR are assumed to be initiated by the failure of plant structures, systems, or components (SSCs). An inoperable UHS is not an initiator of any analyzed events as described in the UFSAR. The impact on the structural integrity of the UHS due to a potential increase water temperature prior to and during the UHS design basis event has been evaluated, and does not increase the probability of the failure of the cooling lake dike. The proposed temperature limit for cooling water supplied to the plant from the CSCS Pond could reduce the commercial capability of the LSCS units; however, it does not result in an increase in the probability of occurrence for any of the events described in the UFSAR.

The basis provided in Regulatory Guide 1.27, “Ultimate Heat Sink for Nuclear Power Plants,” Revision 1, dated March 1974, was employed for the temperature analysis of the LSCS UHS to implement General Design Criteria 2, “Design bases for protection against natural phenomena,” and 44, “Cooling water,” of Appendix A to 10 CFR Part 50. This Regulatory Guide was employed for both the original design and licensing basis of the LSCS UHS and a subsequent evaluation which investigated the potential for changing the average water temperature of the cooling water supplied to the plant from the CSCS Pond from a fixed temperature limit to a limit based on the time of day. The meteorological conditions chosen for the LSCS UHS analysis utilized a 31-day period consisting of the most severe one day, combined with the most severe 30 days based on historical data. The heat loads selected for the UHS analysis considered failure of the cooling lake dike, a loss of offsite power, and a DBA LOCA on one unit and a normal shutdown of the other unit. The LSCS cooling lake is conservatively assumed to be unavailable at the start of the event.

The analysis shows that with an initial UHS temperature less than or equal to the proposed time-of-day-based limit, the required safety-related heat loads can be adequately cooled for 30 days while continuing to ensure safety-related cooling water temperature remains less than the design temperature for LSCS, Units 1 and 2.

Based on the above, it has been demonstrated that the change of the initial temperature limit for cooling water supplied to the plant from the CSCS Pond to less than or equal to a temperature based on the time of day will not impede the ability of the equipment and components cooled by the UHS during a UHS design basis event to perform their safety functions.

There is no impact of this change on LSCS safety analyses including the consequences of all postulated events since all required safety-related equipment continues to perform as designed. The effects of the proposed change on the ability of the UHS to assure that a 30-day supply of water is available considering losses due to evaporation, seepage, and firefighting have been considered. Sufficient inventory remains available to mitigate the design basis event for the LSCS UHS for the required 30-day period.

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

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

Response: No.

The proposed change does not physically alter the operation, testing, or maintenance of any plant SSCs beyond operating with a UHS temperature limit based on the time of day. The proposed change is bounded by existing design analyses. Moreover, the UHS temperature does not initiate accident precursors. The impact of increased UHS temperature can affect the commercial operation of the plant, but the proposed change would not create any accident not considered in the LSCS UFSAR.

This proposed change will not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No alteration in the procedures that ensure the LSCS units remain within analyzed limits is proposed, and no change is being made to procedures relied upon to respond to an off-normal event.

As such, no new failure modes are being introduced. The proposed change does not alter assumptions made in the LSCS safety analysis.

Changing the temperature of cooling water supplied to the plant from the CSCS Pond (i.e., the UHS) as proposed has no impact on plant accident response. The proposed temperature limits do not introduce new failure mechanisms for SSCs. An engineering analysis performed to support the change in temperature of cooling water supplied to the plant from the CSCS Pond provides the basis to conclude that the equipment is adequately designed for operation as proposed.

All systems that are important to safety will continue to be operated and maintained within their design bases, and the proposed change will continue to ensure that all associated systems and components are operated reliably within their design capabilities.

The proposed change will ensure the maximum temperature of the cooling water supplied to the plant during the UHS design basis event remains less than the current safety-related cooling water design temperature for LSCS, Units 1 and 2. Therefore, there is no impact of this change on the LSCS safety analyses including inventory and cooling requirements for safety-related systems using the UHS as their cooling water supply.

All systems will continue to be operated within their design capabilities, no new failure modes are introduced, nor is there any adverse impact on plant equipment; therefore, the proposed change does not result in 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 margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed change does not impact any of these factors. There are no required design changes or equipment performance parameter changes associated with the proposed change. No protection setpoints are affected as a result of this change. The proposed change in the limit for the temperature of cooling water supplied to the plant from the CSCS Pond will not change the operational characteristics of the design of any equipment or system. All accident analysis assumptions and conditions will continue to be met.

   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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Ms. Tamra Domeyer, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Acting Branch Chief: Jeremy S. Bowen.

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

Date of amendment request: March 1, 2013.

Description of amendment request: The proposed amendment will revise the Seabrook Technical Specifications (TSs). The proposed amendment will make administrative changes and corrections to the TSs. The proposed changes delete TS Index and make corrections to TS 3.4.8, "Reactor Coolant System Specific Activity," and TS 6.8.1.6.a, "Core Operating Limits Report."

Basis for proposed NSHC 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes (1) remove the index from the TS, (2) correct an error in the units of activity for 100/E in TS 3.4.8, Reactor Coolant System Specific Activity, and (3) remove an incorrect, non-applicable reference in TS 6.8, Core Operating Limits Report. The proposed changes are all administrative in nature. The administrative changes are not initiators of any accident previously evaluated, and, consequently, the probability and consequences of an accident previously evaluated is not significantly increased.

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

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

   The proposed changes are administrative in nature so no new or different accidents result from the proposed changes. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), a significant change in the method of plant operation, or new operator actions. The changes do not alter assumptions made in the safety analysis.

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

3. The proposed changes do not involve a significant reduction in the margin of safety.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed administrative changes do not involve a change in the method of plant operation, do not affect any accident analyses, and do not relax any safety system settings.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves NSHC.

<u>Attorney for licensee</u>: Mr. James Petro, Managing Attorney, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420.

<u>NRC Branch Chief</u>: Meena Khanna.

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

<u>Date of amendment request</u>: September 28, 2012, as supplemented on November 8, 2012 and December 18, 2012.

<u>Description of amendment request</u>: The proposed amendment requests U.S. Nuclear Regulatory Commission (NRC) approval 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, "Risk-Informed, Performance Based Fire Protection

for Existing Light-Water Nuclear Power Plants." This amendment request also follows the guidance in Nuclear Energy Institute (NEI) 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)." If approved, the PINGP fire protection program would transition to a new Risk-Informed, Performance-Based alternative in accordance with 10 CFR 50.48(c), which incorporates by reference National Fire Protection Association Standard 805 (NFPA 805). The NFPA 805 fire protection program would supersede the current fire protection program licensing basis in accordance with 10 CFR Part 50, Appendix R.

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 Prairie Island Nuclear Generating Plant (PINGP) in accordance with the proposed amendment does not increase the probability or consequences of accidents previously evaluated. Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling evaluations, have been performed to demonstrate that the performance-based requirements of National Fire Protection Association Standard 805 (NFPA 805) have been satisfied. The PINGP Updated Safety Analysis Report (USAR) documents the analyses of design basis accidents (DBAs) at PINGP. The proposed amendment does not adversely affect accident initiators nor alter design assumptions, conditions, or configurations of the facility that would increase the probability or consequences of accidents previously evaluated. Further, the changes to be made for fire hazard protection and mitigation do not adversely affect the ability of structures, systems, and components (SSCs) to perform their design functions, nor do they affect the postulated initiators or assumed failure modes for accidents described and evaluated in the USAR. 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 proposed amendment is to permit PINGP to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 1 of Regulatory Guide (RG) 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 fire protection features (69 FR 33536; June 16, 2004). Engineering analyses, in accordance with NFPA 805, have been performed to demonstrate that the risk-informed, performance-based (RI-PB) requirements per NFPA 805 have been met.

NFPA 805, taken as a whole, provides an acceptable alternative to 10 CFR 50.48(b), satisfies 10 CFR 50.48(a) and General Design Criterion (GDC) 3 of Appendix A to 10 CFR Part 50, and meets the underlying intent of the NRC's existing fire protection regulations and guidance, and provides for defense-in-depth. The goals, performance objectives, and performance criteria specified in Chapter 1 of NFPA 805 ensure that if there are any increases in the net core damage frequency (CDF) or risk associated with this license amendment request (LAR) submittal, the increase will be small and consistent with the Commission's Safety Goal Policy.

Based on this, the implementation of this amendment does not significantly increase the probability of any accident previously evaluated. Equipment required to mitigate an accident remains capable of performing the assumed function(s). The proposed amendment will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated.

Therefore, the consequences of any accident previously evaluated are not significantly increased with the implementation of the proposed amendment.

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 PINGP in accordance with the proposed 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 was included in the evaluation of DBAs documented in the USAR. The proposed change does not alter the requirements or function for systems required during accident conditions. Implementation of the new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 1 of RG 1.205 will not result in new or different accidents.

The proposed amendment does not introduce new or different 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 safely shut down the reactor and maintain it in a safe shutdown condition remain capable of performing their design functions.

The purpose of this amendment is to permit PINGP to adopt a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) and (c) and the guidance in Revision 1 of RG 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 fire protection features (69 FR 33536, June 16, 2004). The requirements in NFPA 805 address only fire protection and the impacts of fire on the plant that have already been evaluated. Based on this, the implementation of this amendment does not create the possibility of a new or different kind of accident from any kind of accident previously evaluated. The proposed amendment does not introduce any new accident scenarios, transient precursors, failure mechanisms, malfunctions, or limiting single failures that could initiate a new accident. There will be no adverse effect or challenges imposed on a safety related system as a result of this proposed amendment.

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

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

Response: No.

Operation of PINGP in accordance with the proposed amendment does not involve a significant reduction in a margin of safety. The proposed amendment 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 amendment does not adversely affect existing plant safety margins or the reliability of equipment assumed to mitigate accidents in the USAR. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition remain capable of performing their design function.

The purpose of this amendment is to permit PINGP to adopt a new fire protection licensing basis which complies with the requirements in

> 10 CFR 50.48(a) and (c) and the guidance in Revision 1 of RG 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 fire protection features (69 FR 33536; June 16, 2004). Engineering analyses, which may include engineering evaluations, probabilistic safety assessments, and fire modeling evaluations, have been performed to demonstrate that the performance-based methods do not result in a significant reduction in a margin of safety.
>
> Based on this, the implementation of this amendment does not significantly reduce a margin of safety. The proposed changes are evaluated to ensure that the risk and safety margins are kept within acceptable limits.
>
> Therefore, the transition to NFPA 805 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 requests involve no significant hazards consideration.

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

NRC Branch Chief: Robert D. Carlson.

PPL Susquehanna, LLC, Docket Nos. 50-387 and 50-388, Susquehanna Steam Electric Station, (SSES) Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment requests: December 19, 2012.

Description of amendment requests: The proposed amendments would modify the SSES Unit 1 and SSES Unit 2 Technical Specifications (TS) Section 2.1.1 to reflect a revised Low Pressure Safety Limit. The change to TS Section 2.1.1 became necessary as a result of General Electric

(GE) PART 21 REPORT, SC05-03, “Potential to Exceed Low Pressure Technical Specification Safety Limit.”

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

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

   Response: No.

   The proposed amendment changes the low pressure safety limit in Technical Specification (TS) 2.1.1 from 785 psig [pounds per square inch gauge] to 557 psig based on the capabilities of the current critical power correlation used by Susquehanna (SPCB). The SPCB correlation is approved for CPR [critical power ratio] calculations by the NRC for reactor pressures > 571.4 psia [pounds per square inch absolute] and is listed as an approved analytical method in TS 5.6.5.b.

   The proposed changes will not alter existing Final Safety Analysis Report (FSAR) design basis accident analysis assumptions, add any accident initiators, or affect the function of the plant safety-related structures, systems, or components (SSCs) as to how they are operated, maintained, modified, tested, or inspected.

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

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

   Response: No.

   The change to the Low Pressure Safety Limits does not result in the need for any new or different FSAR design basis accident analysis. The inclusion does not introduce new equipment that could create a new or different kind of accident, and no new equipment failure modes are created. In addition, the proposed change does not affect the function of any safety-related SSC as to how they are operated, maintained, modified, tested or inspected. As a result, no new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed amendment.

Therefore, the proposed amendment does not create a possibility for an accident of a new or different type than those previously evaluated.

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

Response: No.

The margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation to the public. Evaluation of the 10 CFR Part 21, "Reporting of Defects and Noncompliance" issue that identified the need for the proposed change determined that there was no decrease in the safety margin and therefore no threat to fuel cladding integrity. The proposed changes to the Low Pressure Safety Limits would not alter the way safety-related SSCs function and would not alter the way PPL Susquehanna Units 1 and 2 are operated. The proposed changes to the safety limit are within the capabilities of the existing NRC approved CPR correlation and ensure valid CPR calculations for the Anticipated Operational Occurrences (AOOs) defined in the FSAR. The proposed amendment would have no impact on the structural integrity of the fuel cladding, reactor coolant pressure boundary, or containment structure. Based on the above considerations, the proposed amendment would not degrade the confidence in the ability of the fission product barriers to limit the level of radiation to the public.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101-1179.

NRC Branch Chief: Meena K. Khanna.

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

Date of amendment request: July 17, 2012, as supplemented on January 28, 2013.

Description of amendment request: The proposed amendments would revise Salem Technical Specifications (TS) 3.7.6.1 (Unit 1) and 3.7.6 (Unit 2), "Control Room Emergency Air Conditioning System," to eliminate the separate action statements for securing an inoperable Control Area Air Conditioning System and Control Room Emergency Air Conditioning System isolation damper in the closed position and entering the actions for an inoperable control room envelope boundary.

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 Control Room Emergency Air Conditioning System (CREACS) is not an initiator of or a precursor to any accident or transient. The CREACS system is in standby during normal operation and initiates in the event of a safety injection signal or control room radiation monitoring actuation in response to a design basis accident to pressurize the Control Room Envelope (CRE) and provide filtration of the CRE atmosphere to maintain the control room operator doses within the limits of General Design Criteria (GDC) 19. The system also operates in recirculation mode to mitigate the consequences of a fire or toxic gas release that occurs outside of the CRE.

   The design of plant equipment is not being modified by the proposed amendment. The elimination of the action to secure the isolation dampers between the normal Control Area Air Conditioning System (CAACS) and the CREACS when these dampers are inoperable and entering the actions for the inoperable control room boundary will ensure operation of the plant within the limits of the radiological, smoke and chemical hazard analyses. The intent of the original action for securing the inoperable isolation damper in the closed position was to maintain the boundary of the CRE. The actions for an inoperable control room boundary ensure that mitigating actions are implemented that maintain the CRE boundary within the limits of the radiological, smoke and chemical hazard analyses.

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

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

   Response: No.

   The proposed changes to the TS to implement the actions for an inoperable control room boundary when a normal CAACS and CREACS isolation damper is inoperable do not introduce any new accident precursors and do not involve any physical plant alterations or changes in the methods governing normal plant operation that could initiate a new or different kind of accident. The proposed amendment does not alter the function of the system to initiate and pressurize the control room envelope in the event of a DBA nor alter the ability to initiate CREACS in the recirculation mode in response to a fire or chemical release that occurs outside of the CRE.

   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.

   Margin of safety is related to the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. The proposed amendment does not alter setpoints or limits established or assumed by the accident analyses. The control room envelope is considered a barrier for the control room operators during a design basis accident radiological release and a barrier in the event of a fire or chemical hazard that occurs outside of the CRE. Implementing the actions for an inoperable control room boundary in the event of an inoperable isolation damper between the normal CAACS and CREACS ensure operation of the plant within the limits of the radiological, smoke and chemical hazard analysis. The actions for an inoperable control room boundary ensure that mitigating actions are implemented that maintain the CRE boundary within the limits of the radiological, smoke

> and chemical hazard analyses. Therefore the plant will continue to be operated consistent with the plant safety analyses.
>
> 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 requests involve no significant hazards consideration.

Attorney for licensee: Jeffrie J. Keenan, PSEG Nuclear LLC - N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Meena K. Khanna.

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

Date of amendment request: March 13, 2013.

Description of amendment request: The proposed change would amend Combined License Nos.: NPF-93 and NPF-94 for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 in regard to the Chemical and Volume Control System (CVS) by: (1) providing a spring-assisted check valve around the air-operated Reactor coolant System (RCS) Purification Return Line Stop Check Valve, (2) replacing the CVS zinc addition inboard containment isolation lift check valve with an air-operated globe valve and a thermal relief valve and (3) separating the zinc and hydrogen injection paths and relocate the zinc injection path.

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

<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 changes to provide a spring-assisted check valve located in the bypass line around the makeup stop check valve would continue to meet the existing design functions because the ASME Boiler and Pressure Vessel Code (ASME Code) Section III valves will maintain the flow isolation design function and preserve the Reactor Coolant System (RCS) pressure boundary safety function. The replacement of the Chemical and Volume Control System (CVS) zinc addition inboard containment isolation lift check valve with an air operated globe valve and addition of a pressure relief valve would continue to meet the containment isolation and RCS pressure boundary design functions because the replacement valves will be designed, analyzed, tested and qualified, including seismic qualification, to ASME Code Section III requirements. Separating the zinc and hydrogen injection paths and relocating the zinc injection point would continue to meet containment boundary requirements, including containment isolation and in-service testing, and preserve the RCS pressure boundary safety functions because the revised containment isolation configuration is consistent with those described in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 55, and the additional valves and piping will be qualified to ASME Code Section III. Because the proposed CVS changes would preserve the CVS safety-related design functions, the probability of an accident previously evaluated is not affected.

   The CVS safety functions have been preserved, because the proposed CVS configuration changes, including revised valve types, will perform the same safety functions as the current design. The proposed CVS configuration changes would neither impact any accident source term parameter or fission product barrier nor affect radiological dose consequence analysis.

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

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

   Response: No.

   The additional containment penetration is similar in form, fit, and function to the CVS combined zinc/hydrogen containment penetration that is currently described in the Updated Final Safety Analysis Report. Because the CVS changes use valve types, piping, and a containment penetration consistent with those already described in the Updated Final Safety Analysis Report, no new failure modes or equipment failure initiators are introduced by these changes. Accordingly, the proposed changes do not create any new malfunctions, failure mechanisms, or accident initiators.

   Therefore, the proposed amendment 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 containment isolation and pressure relief functions would not be changed by this activity and are consistent with the existing design. The proposed CVS containment penetration is similar in form, fit, and function to existing CVS combined zinc/hydrogen containment penetration and, therefore, does not affect containment or its ability to perform its design function. The addition of these CVS components, including piping, a spring-assisted check valve, an air-operated containment isolation valve, a thermal relief valve and the additional CVS containment penetration do not impact a design basis or safety limit. Because the CVS design functions of controlling the RCS oxygen concentration, reducing radiation fields, containment isolation and overpressure protection within existing limits are not changed by this activity and are bounded by the existing design, there is no change to any current margin of safety.

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

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

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

NRC Acting Branch Chief: Lawrence Burkhart.

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

Date of amendment request: February 15, 2013.

Description of amendment request: The proposed change would amend Combined Licenses Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 by departing from the plant-specific design control document Tier 2* material by revising reference document APP-OCS-GEH-320, "AP1000 Human Factors Engineering Integrated System Validation Plan" from Revision D to Revision 2. APP-OCS-GEH-320 is incorporated by reference in the updated final safety analysis report (UFSAR) as a means to implement the activities associated with the human factors engineering verification and validation.

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 Integrated System Validation (ISV) provides a comprehensive human performance-based assessment of the design of the AP1000 Human-System Interface (HSI) resources, based on their realistic operation within a simulator-driven Main Control Room (MCR). The ISV is part of the overall AP1000 Human Factors Engineering (HFE) program. The changes are to the ISV Plan to clarify the scope and amend the details of the methodology. The ISV Plan is needed to perform, in the simulator, the scenarios described in the document. The functions and tasks allocated to plant personnel can still be accomplished after the proposed changes. The performance of the tests governed by the ISV Plan provides additional assurances that the operators can appropriately respond to plant transients. The ISV Plan does not affect the plant itself. Changing the ISV Plan does not affect prevention and mitigation of abnormal events, e.g., accidents, anticipated operational occurrences, earthquakes, floods and turbine missiles, or their safety or design analyses. No safety-related structure, system, component (SSC) or function is adversely affected. The changes do not involve nor interface with any SSC accident initiator or initiating sequence of events, and thus, the probabilities of the accidents evaluated in the UFSAR are not affected. Because the changes do not involve any safety-related SSC or function used to mitigate an accident, the consequences of the accidents evaluated in the UFSAR 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 changes to the ISV Plan affect the testing and validation of the Main Control Room and Human System Interface using a plant simulator. Therefore, the changes do not affect the safety-related equipment itself, nor do they affect equipment which, if it failed, could initiate an accident or a failure of a fission product barrier. No analysis is adversely affected. No system or design function or equipment qualification will be adversely affected by the changes. This activity will not allow for a new fission product release path, nor will it result in a new fission product barrier failure mode, nor create a new sequence of events that would result in significant fuel cladding failures. In addition, the changes do not result in a new failure mode, malfunction or sequence of events that could affect safety or safety-related equipment.

   Therefore, this activity does not create the possibility of a new or different kind of accident than any accident previously evaluated.

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

   Response: No.

   The changes to the ISV Plan affect the testing and validation of the Main Control Room and Human System Interface using a plant simulator. Therefore, the changes do not affect the assessments or the plant itself. These changes do not affect safety-related equipment or equipment whose failure could initiate an accident, nor does it adversely interface

> with safety-related equipment or fission product barriers. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the requested change.
>
> Therefore, there is no 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.

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NRC Acting Branch Chief: Lawrence Burkhart.

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

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

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

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

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

<u>Exelon Generation Company, LLC, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania</u>

<u>Date of application for amendments</u>: November 14, 2012.

<u>Brief description of amendments</u>: The amendments relocate the Technical Specification (TS) requirements for motor-operated valve thermal overload protection from the TSs to the Technical Requirements Manual.

<u>Date of issuance</u>: March 19, 2013.

<u>Effective date</u>: As of the date of issuance, to be implemented within 60 days.

<u>Amendments Nos.</u>: 209 and 170.

<u>Facility Operating License Nos. NPF-39 and NPF-85</u>: The amendments revised the License and TSs.

<u>Date of initial notice in *Federal Register*</u>: January 8, 2013 (78 FR 1270).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 2013.

<u>No significant hazards consideration comments received</u>: No.

<u>FirstEnergy Nuclear Operating Company, et al., Docket No. 50-440, Perry Nuclear Power Plant (PNPP), Unit 1, Lake County, Ohio</u>

<u>Date of application for amendment</u>: July 3, 2012, supplemented by letter dated January 7, 2013.

<u>Brief description of amendment</u>: The proposed amendment would modify PNPP's Technical Specifications (TS) 3.8.1, "AC [alternating current] Sources - Operating." Specifically, the proposed amendment will modify nine surveillance requirements (SRs) by excluding Division 3

from the current mode restrictions, thus allowing performance of the subject SRs in any mode of plant operation. The proposed amendment also deletes expired TS 3.8.1 provisions regarding use of a delayed access circuit.

Date of issuance: March 5, 2013.

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

Amendment No.: 162.

Facility Operating License No. NPF-58: This amendment revised the Technical Specifications and License.

Date of initial notice in *Federal Register*: November 13, 2012 (77 FR 67682). The January 7, 2013 supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2013.

No significant hazards consideration comments received: No.

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: January 18, 2013.

Brief description of amendment: The proposed amendment would depart from VEGP Units 3 and 4 plant-specific Design Control Document (DCD) Tier 2* material incorporated into the Updated Final Safety Analysis Report (UFSAR) by revising the structural criteria code for anchoring of headed shear reinforcement bar within the nuclear island basemat.

Date of issuance: March 1, 2013.

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

Amendment No.: Unit 3--5, and Unit 4--5.

Facility Combined Licenses No. NPF-91 and NPF-92: Amendment revised the Facility Combined Licenses.

Date of initial notice in *Federal Register*: January 29, 2013 (78 FR 6142).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 25th day of March 2013.

FOR THE NUCLEAR REGULATORY COMMISSION,

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

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