[7590-01-P]

UNITED STATES NUCLEAR REGULATORY COMMISSION

BIWEEKLY NOTICE

APPLICATIONS AND AMENDMENTS TO FACILITY OPERATING LICENSES

INVOLVING NO SIGNIFICANT HAZARDS CONSIDERATIONS

[NRC-2010-0297]

I. 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 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 August 26, 2010, to September 8, 2010. The last biweekly notice was published on September 7, 2010 (75 FR 54390-54400).

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING 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), Section 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.

Written comments may be submitted by mail to the Chief, Rules, Announcements and Directives Branch (RADB), TWB-05-B01M, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this *Federal Register* notice. Written comments may also be faxed to the RADB at 301-492-3446. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

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. 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 Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, 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, 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 ten (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 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 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 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 EIE, users will be required to install a Web browser plug-in from the NRC 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 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 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 e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (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 NRC's electronic hearing docket which is available to the public at http://ehd.nrc.gov/EHD_Proceeding/home.asp, 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 Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, 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 PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: July 20, 2010.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.1.2, "Emergency Feedwater System," to clarify the acceptability of transitioning from Mode 4 to Mode 3 with the turbine-driven emergency feedwater (EFW) pump inoperable but available. This proposal would grant an exception to TS LCO 3.0.4 and Surveillance Requirement 4.0.4 allowing entry into operational Mode 3 with TS LCO equipment, the turbine-driven EFW pump, associated with a shutdown action inoperable.

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 addition of an exception to TS LCO 3.0.4 for entry into Mode 3 during a plant startup for the turbine-driven EFW pump for a plant condition when the turbine driven EFW pump would be unable to complete its post maintenance activities (i.e. dynamic final calibration of the governor valve speed control unit governor control system) due to insufficient steam pressure in the steam generator secondary side and then to complete the quarterly IST [Inservice Testing] and 18 month EFAS [Engineered Safety Features Actuation System] SR [Surveillance Requirement] within the allowance of the delay of the respective SR is administrative in nature.

   This change will clarify that the turbine-driven EFW pump is not required to fully demonstrate operability (i.e. be inoperable pending completion of the quarterly IST and 18 month EFAS SR) during plant startup prior to entry into Mode 3 under the conditions and for the period as provided in the quarterly IST and 18 month EFAS SR as granted by the NRC [Nuclear Regulatory Commission] in Reference 7.1 [NRC letter to Waterford 3 dated October 4, 2001, Waterford Steam Electric Station - Unit 3, Issuance of Amendment RE: Emergency Feedwater System (TAC No MB2010), Agencywide Documents Access and Management System (ADAMS) Accession No. ML012840538]. When the plant enters Mode 3 during plant startup, the turbine-driven EFW pump is available (i.e., there is a reasonable expectation that once sufficient steam pressure is available to the turbine-driven EFW pump turbine, it will be

able to successfully complete the quarterly IST and 18 month EFAS surveillance requirements to fully demonstrate operability).

Prior to entry into Mode 2, surveillance requirement testing of various combinations of EFW pumps and valves will ensure ALL required EFW system flow paths and equipment (which includes the turbine-driven EFW pump) are demonstrated operable before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown.

Since the two motor-driven EFW pumps are required to be operable when entering Modes 3 from Mode 4, then for the worst case postulated accident scenario during plant startup, with the turbine-driven EFW pump considered inoperable but available (utilizing the exception to TS LCO 3.0.4 as tied to the quarterly IST and 18 month EFAS SR for fully demonstrating operability of the turbine-driven EFW pump), the EFW System safety function of achieving shutdown cooling entry conditions would be met.

This request is merely a clarification and does not present any change to equipment operation, design or practices. The proposed clarification is not an accident initiator and will not adversely affect plant safety functions. The EFW System capability to provide its specified function of being able to achieve shutdown cooling entry conditions of the Reactor Coolant [S]ystem is unchanged by this clarification.

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

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

   Response: No.

   The proposed addition of an exception to TS LCO 3.0.4 for entry into Mode 3 during a plant startup for the turbine-driven EFW pump for a plant condition when the turbine-driven EFW pump would be unable to complete its post maintenance activities (i.e. dynamic final calibration of the governor valve speed control unit governor control system) due to insufficient steam pressure in the steam generator secondary side and then to complete the quarterly IST and 18 month EFAS SR within the allowance of the delay of the respective SR is administrative in nature.

   This change will clarify that the turbine-driven EFW pump is not required to fully demonstrate operability (i.e. be inoperable pending completion of the quarterly IST and 18 month EFAS SR) during plant startup prior to entry into Mode 3 under the conditions and for the period as provided in the quarterly IST and 18 month EFAS SR as granted by the NRC in Reference 7.1. When the plant enters Mode 3 during plant startup, the turbine-driven EFW pump is available (i.e. there is a reasonable expectation that once sufficient steam

pressure is available to the turbine-driven EFW pump turbine, it will be able to successfully complete the quarterly IST and 18 month EFAS surveillance requirements to fully demonstrate operability).

Prior to entry into Mode 2, surveillance requirement testing of various combinations of EFW pumps and valves will ensure ALL required EFW system flow paths and equipment (which includes the turbine-driven EFW pump) are demonstrated operable before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown.

The addition of this exception to TS LCO 3.0.4 for the turbine-driven EFW pump introduces no new mode of plant operation and does not alter the EFW System functional capability. The scope of this proposed change does not establish a potential new accident precursor. This proposed change will not change the design, configuration or method of operation of the EFW System. No new possibility for an accident is introduced by the proposed clarification.

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 addition of an exception to TS LCO 3.0.4 for entry into Mode 3 during a plant startup for the turbine-driven EFW pump for a plant condition when the turbine-driven EFW pump would be unable to complete its post maintenance activities (i.e. dynamic final calibration of the governor valve speed control unit governor control system) due to insufficient steam pressure in the steam generator secondary side and then to complete the quarterly IST and 18 month EFAS SR within the allowance of the delay of the respective SR is administrative in nature.
   This change will clarify that the turbine-driven EFW pump is not required to fully demonstrate operability (i.e. be inoperable pending completion of the quarterly IST and 18 month EFAS SR) during plant startup when entering Mode 3 under the conditions and for the period as provided in the quarterly IST and 18 month EFAS SR as granted by the NRC in Reference 7.1. When the plant enters Mode 3 during plant startup, the turbine-driven EFW pump is available (i.e. there is a reasonable expectation that once sufficient steam pressure is available to the turbine-driven EFW pump turbine, it will be able to successfully complete the quarterly IST and 18 month EFAS surveillance requirements to fully demonstrate operability).

   Prior to entry into Mode 2, surveillance requirement testing of various combinations of EFW pumps and valves will ensure ALL required EFW system flow paths and equipment (which includes the turbine-driven EFW

> pump) are demonstrated operable before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown.
>
> The proposed clarification does not adversely affect Emergency Feedwater equipment operating practices. The EFW System has the same capabilities as before to mitigate accidents. Surveillance requirements are not reduced by the proposed change. The EFW System capability to provide its specified function of being able to achieve shutdown cooling entry conditions of the Reactor Coolant System following a worst case postulated accident is unchanged by this clarification.
>
> 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: Joseph A. Aluise, Associate General Counsel - Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

NextEra Energy Point Beach, LLC (the licensee), Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant (PBNP), Units 1 and 2, Town of Two Creeks, Manitowac County, Wisconsin

Date of amendment request: April 7, 2009, as supplemented by letters dated June 17, September 11, November 20, November 30, and December 8 of 2009; and February 11, February 25, April 22, April 30, July 21, July 28, and August 2 of 2010.

Description of amendment request: The proposed amendment would revise Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation setpoints for the PBNP, Units 1 and 2. The revised Technical Specification (TS) allowable values are specified in Tables 3.3.1-1 and 3.3.2-1 for RPS and ESFAS, respectively. These changes were originally included as part of the April 7, 2009, extended power uprate (EPU)

license amendment request, but subsequently divided into a separate licensing action for independent technical review. The proposed changes include both EPU and non-EPU related changes.

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

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

   Response: No.

   The proposed changes to the TSs will ensure that the results of previously evaluated accidents at the uprated conditions remain within the acceptance criteria. The proposed RPS and ESFAS setpoint changes provide appropriate values for operation at EPU conditions. The revised TS allowable values have been calculated to account for new EPU analytical limits, instrument uncertainties, and instrument drift. The proposed RPS and ESFAS setpoint changes are considered in the safety analysis for the affected RPS and ESFAS functions, and do not significantly increase the probability or consequences of the accidents previously evaluated and the setpoint changes considered in the safety analysis continue to meet the applicable acceptance criteria. The safety analyses for these accidents have been performed at the EPU power level and demonstrated acceptable results.

   The proposed changes will ensure that the instruments actuate as assumed to mitigate accidents previously evaluated. The proposed changes will not significantly affect accident initiators or precursors and will not alter or prevent the ability of systems, structures, or components from performing the intended safety function to meet the applicable acceptance limits for the accidents and events.

   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 change does not involve a physical alteration of the plant or change the methods governing normal plant operation. The change does not alter assumptions made in the safety analyses, but ensures that the instruments behave as assumed in the accident analysis. The proposed change is

consistent with the safety analysis assumptions. The proposed RPS and ESFAS Limiting Safety System Setting (LSSS) changes do not create the possibility of a new or different type of accident due to operation at EPU conditions. The revised TS LSSS values have been calculated to account for new EPU analytical limits and known instrument uncertainties. The proposed RPS and ESFAS setpoint changes are used in the safety analysis for the affected RPS and ESFAS functions, and do not significantly affect these accidents or the applicable acceptance criteria.

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

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

   Response: No.

   The proposed changes clarify the TS requirements for instrumentation to ensure that the automatic protection action will correct the abnormal situation before a safety limit is exceeded. The proposed change also revises the TSs to enhance the controls used to maintain the variables and systems within the prescribed operating ranges, in order to ensure that automatic protection actions occur to initiate the operation of systems and components important to safety as assumed in the accident analysis. No change is made to the accident analysis assumptions.

   The proposed changes to the RPS and ESFAS setpoint TSs provide adequate margin such that PBNP Units 1 and 2 can be operated in a safe manner at EPU conditions. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. All systems, structures and components previously assumed for the mitigation of an event remain capable of fulfilling their intended function. The proposed changes will not have any significant effect on the margin of safety.

   Therefore, the proposed 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 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: William Blair, Senior Attorney, NextEra Energy Point Beach, LLC, P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Robert J. Pascarelli.

NextEra Energy Point Beach, LLC (the licensee), Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowac County, Wisconsin

Date of amendment request: April 7, 2009, as supplemented by letters dated June 17 (two letters), September 11, September 25, October 9, November 20 (two letters), November 21 (two letters), November 30, December 8, and December 16 of 2009; and January 7, January 8, January 22, February 11, February 25, March 3, April 15, April 22, July 8, July 28, August 2, August 9, and August 24 of 2010.

Description of amendment request: The proposed amendment would change the auxiliary feedwater (AFW) system design and Technical Specifications (TS) 3.7.5, "Auxiliary Feedwater (AFW)," and TS 3.7.6, "Condensate Storage Tank (CST)," resulting from 1) modifications to the AFW system to support requirements for transients and other accidents at extended power uprate (EPU) conditions; 2) installation of main feedwater isolation valves to support accident mitigation by ensuring that containment pressure does not exceed safety analysis limits; 3) automatic AFW switchover from a CST suction source to a safety-related Service Water (SW) source; and 4) setpoint changes supporting the aforementioned physical modifications. These changes were originally included as part of the April 7, 2009, EPU license amendment request, but subsequently divided into a separate licensing action for independent technical review. The upgrades and modifications to the AFW system are being installed to provide additional capacity and reliability for the system. Although the proposed changes are also designed to support the requirements for transients and other accidents at EPU conditions, the proposed changes for this amendment are being evaluated using the current licensing basis.

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. The NRC staff performed its own analysis, which is presented below:

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

   Response: No.

   The design functions of the AFW system will not be altered by the proposed change. The AFW system will continue to perform its original intended design function, mitigating the consequences of accidents previously evaluated. The proposed changes will not significantly affect accident initiators or precursors. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed modifications.

   Implementation of the new AFW system design and the proposed changes to TS 3.7.5 was evaluated against the current analysis of record for the current licensed power level at PBNP, Units 1 and 2. The current analyses remain applicable or are unaffected by implementation of the new AFW system and associated TS changes, with the exception of the steam line break containment response and steam generator tube rupture (SGTR) radiological consequences. These two accidents were reanalyzed with the current licensing basis for the AFW modifications and the results were acceptable with the revised minimum and maximum AFW flow rates and pump start timing.

   Therefore, the consequences of accidents previously evaluated for the current licensed power level are not significantly increased.

   A proposed change to TS 3.7.6 changes the surveillance requirement (SR) for minimum CST water inventory to be maintained to supply AFW pump suction in the event of a Station Blackout, when the safety-related AFW suction source from the SW system is not available. The proposed TS 3.7.6 SR increases the current minimum required inventory to account for the increased flow rates from the new AFW system design, suction piping losses, instrument uncertainties, vortex prevention, net positive suction head (NPSH) requirements, and the suction of the AFW pumps under various combinations of CST and plant units in operation. This change to the minimum required CST level inventory will not increase the probability or consequences of previously evaluated accidents.

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

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

   Response: No.

The proposed change does not introduce a new mode of plant operation. The proposed changes involving the AFW system do not significantly alter any design basis accident or event response. The proposed changes will not significantly affect accident initiators or precursors. The AFW system will continue to perform its design function. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed modifications. All systems, structures, and components previously assumed for the mitigation of an event remain capable of fulfilling their intended design function. The new AFW system design and proposed changes to TS 3.7.5 and the proposed increase in CST inventory in TS 3.7.6 do not create the possibility of a new or different kind of accident or event.

As previously discussed, implementation of the new AFW system design and the proposed changes to TS 3.7.5 was evaluated against the current analysis of record for the current licensed power level at PBNP, Units 1 and 2. The current analyses remain applicable or are unaffected by implementation of the new AFW system and associated TS changes, with the exception of the steam line break containment response and steam generator tube rupture (SGTR) radiological consequences. These two accidents were reanalyzed with the current licensing basis for the AFW modifications and the results are acceptable with the revised minimum and maximum AFW flow rates and pump start timing. The AFW system design change, the changes to TS .3.7.5, and the increase in required CST inventory established in TS 3.7.6, are not significant accident initiators or precursor and will not create the possibility of a new or different kind of accident.

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

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

   Response: No.

   The upgrade to the AFW system is being made to support requirements for transients and other accidents at EPU conditions. This modification to the AFW system will provide additional capacity and reliability for the system. As such, the proposed amendment does not involve a significant reduction in safety.

   The analyses and evaluations of the Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) systems based on completion of the required modifications, confirm that the systems and components will function as designed and demonstrate that the NSSS and BOP systems and components meet all applicable design and licensing requirements at the uprated power level.

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

Based on the above 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: William Blair, Senior Attorney, NextEra Energy Point Beach, LLC,.P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Robert J. Pascarelli.

NextEra Energy Point Beach, LLC (the licensee), Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowac County, Wisconsin

Date of amendment request: June 1, 2010, as supplemented by letter dated July 9, 2010.

Description of amendment request: The proposed amendment consists of revising the current license basis regarding a postulated reactor vessel head (RVH) drop event to conform to the NRC-endorsed guidance of Nuclear Energy Institute (NEI) 08-05, “Industry Initiative on Control of Heavy Loads,” Revision 0. The proposed change to the license basis will revise Chapter 14.3.6, “Reactor Vessel Head Drop Event,” of the Final Safety Analysis Report. The current license basis assumes failure of the reactor coolant system (RCS) boundary caused by the predicted maximum downward displacement of the reactor vessel which would sever all 36 bottom-mounted instrument (BMI) conduit tubes. The new analysis demonstrates that a postulated RVH drop would not result in a loss of RCS inventory caused by an RCS boundary failure, since the BMI conduits would remain intact.

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 is limited in scope to a postulated RVH drop and the administrative controls in place, which limit the height of the RVH lift, ensuring an actual drop is bounded by the analyses of record.

   Incorporation of the analysis performed in accordance with NRC-approved guidance, which demonstrates bottom-mounted instrumentation (BMI) conduits will not sever following a postulated RVH drop, does not increase the probability or consequences of a previously evaluated accident. The evaluation, in fact, demonstrates that if the postulated RVH drop occurred, the consequences would be significantly less than are now assumed because the ability to maintain a coolable geometry in the core has not been compromised. In accordance with NRC-endorsed methodology contained in NEI 08-05, which states, “Previous evaluations have indicated that the consequences of impacts between the upper vessel internals and the fuel were not significant with respect to public health and safety,” a revised radiological analysis was not performed.

   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 amendment is limited in scope to a postulated RVH drop and the administrative controls in place, which limit the height of the reactor RVH lift, ensuring an actual drop is bounded by the analysis of record

   Incorporation of the analysis performed in accordance with NRC-approved guidance, which demonstrates BMI conduits will not sever following a postulated RVH drop, does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment does not: (1) operate equipment in alignments or in a manner different form that previously evaluated in the FSAR; (2) install, remove or modify equipment important to safety; or (3) introduce new failure modes or effects for any existing system, structure or component.

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

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

Response: No.

The proposed amendment is limited in scope to a postulated RVH drop and the administrative controls in place, which limit the height of the reactor RVH lift, ensuring an actual drop is bounded by the analysis of record.

Incorporation of the analysis performed in accordance with NRC-approved guidance, which demonstrates BMI conduits will not sever following a postulated RVH drop, does not involve a significant reduction in the margin of safety. The evaluation, in fact, demonstrates that if the postulated RVH drop occurred, the consequences would be significantly less than are now assumed because the ability to maintain a coolable geometry in the core has not been compromised.

Therefore, the proposed 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 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: William Blair, Senior Attorney, NextEra Energy Point Beach, LLC, P. O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Robert J. Pascarelli.

Northern States Power Company - Minnesota, Docket No. 50-263, Monticello Nuclear Generating Plant (MNGP), Wright County, Minnesota

Date of amendment request: January 21, 2010.

Description of amendment request: The licensee proposed to amend the MNGP Technical Specifications to allow operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) expanded domain. The licensee stated that the Nuclear Regulatory Commission (NRC) had previously approved various aspects of the MELLLA+ methodology, but that the current application is the first plant-specific use of such methodology. The amendment would include changes to the Technical Specifications to: (1) prohibit the use of the MELLLA+

expanded operating domain when in single loop operation; (2) change the allowable value for Average Power Range Monitor (APRM)-Simulated Thermal Power - High; (3) eliminate an unnecessary surveillance requirement; (4) require certain content in the Core Operating Limits Report. Approval of this amendment would allow the licensee to implement operational changes to provide increased operational flexibility for power maneuvering, to compensate for fuel depletion, and to maintain efficient power distribution in the reactor core without the need for more frequent rod pattern changes. MELLLA+ would increase the operating range to the Extended Power Uprate rated thermal power at 80 percent flow; thus creating a 20 percent flow-control window. By operating in the MELLLA+ domain, a significantly lower number of control rod movements will be required than in the present operating domain. This would represent a significant improvement in operating flexibility. It also provides safer operation, because reducing the number of control rod manipulations would minimize the likelihood of fuel failures, and reduce the likelihood of accidents initiated by reactor maneuvers.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC). The licensee's NSHC analysis is reproduced below:

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

   Response: No.

   The probability (frequency of occurrence) of [d]esign [b]asis [a]ccidents occurring is not affected by the MELLLA+ operating domain, because MNGP continues to comply with the regulatory and design basis criteria established for plant equipment. Further, a probabilistic risk assessment demonstrates that the calculated core damage frequencies do not significantly change due to the MELLLA+.

   There is no change in consequences of postulated accidents, when operating in the MELLLA+ operating domain compared to the operating domain

previously evaluated. The results of accident evaluations remain within the NRC[-]approved acceptance limits.

The spectrum of postulated transients has been investigated and is shown to meet the plant's currently licensed regulatory criteria. In the area of fuel and core design, for example, the Safety Limit Minimum Critical Power Ratio (SLMCPR) is still met. Continued compliance with the SLMCPR will be confirmed on a cycle[-]specific basis consistent with the criteria accepted by the NRC.

Challenges to the [r]eactor [c]oolant [p]ressure [b]oundary were evaluated for the MELLLA+ operating domain conditions (pressure, temperature, flow, and radiation) and were found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment were evaluated and the containment and its associated cooling systems continue to meet the current licensing basis. The calculated post[-]LOCA [loss-of-coolant accident] suppression pool temperature remains acceptable.

Therefore, the proposed changes do 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.

   Equipment that could be affected by the MELLLA+ operating domain has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. The MELLLA+ operating domain uses developed technology and applies it within the capabilities of existing plant safety-related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). No new accident or event precursor has been identified.

   The-MNGP TS require revision to implement the MELLLA+ operating domain. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated.

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

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

   Response: No.

> The MELLLA+ operating domain affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were evaluated for the MELLLA+ operating domain conditions. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on affected structures, systems and components, including the reactor coolant pressure boundary, will remain within their design allowables for design[-]basis event categories. No NRC acceptance criterion is exceeded. Because the MNGP configuration and responses to transients and postulated accidents do not result in exceeding the presently approved NRC acceptance' limits, 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 proposed amendment involves no significant hazards consideration.

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

NRC Branch Chief: Robert J. Pascarelli.

Northern States Power Company - Minnesota, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment request: June 14, 2010.

Description of amendment request: The proposed amendments would revise the Technical Specifications to allow the use of a dedicated on-line core power distribution monitoring system (PDMS) to enhance surveillance of core thermal limits. The PDMS to be used at Prairie Island Nuclear Generating Plant, Units 1 and 2, is the Westinghouse proprietary core analysis system called the Best Estimate Analyzer for Core Operations - Nuclear (BEACON™).

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 PDMS performs continuous core power distribution monitoring with data input from existing plant instrumentation. The system passively supports Technical Specification (TS) surveillances which ensure that core power distribution is within the same limits that are currently prescribed. Further, the proposed TS Actions are comparable to existing operator actions such that no new plant configurations are prompted by the proposed change. The system's physical interface with plant equipment is limited to an electronic link from a new workstation to the plant process computer. The system is passive in that it provides no control or alarm functions, and does not promote any new plant configuration which would affect the initiation, probability, or consequences of a previously-evaluated accident. Continuous on-line core monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This system performance may result in an earlier determination of an adverse core condition and more time for operator action, thus reducing the probability of an accident occurrence and reduced consequences should a previously-evaluated accident occur.

   By virtue of its inherently passive surveillance function and limited interface with plant systems, structures, or components, the proposed changes will not result in any additional challenges to plant equipment that could increase the probability or occurrence of any previously-evaluated accident. Further, the proposed changes will ensure conformance to the same core power distribution limits that form the basis for initial conditions of previously evaluated accidents. Thereby, the proposed changes will not affect the consequences of any previously-evaluated accident.

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

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

   Response: No.

   The system's physical interface with plant equipment is limited to an electronic link from a new workstation to the plant process computer. The system is passive in that it provides no control or alarm functions, and the proposed changes (including operator actions prescribed by the proposed TS) do not promote any new plant configuration which would create the possibility for an accident of a new or different type.

The NRC previously evaluated the effects of using the PDMS to monitor core power distribution parameters and determined that all design standards and applicable safety criteria limits are met. The Technical Specifications will continue to require operation within the required core operating limits, and appropriate actions will continue to be taken when or if limits are exceeded. Thus, the reactor core will continue to be operated within its reference bounds of design such that an accident of a new or different type is not credible.

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

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

   Response: No.

   No margin of safety is adversely affected by the implementation of the PDMS. The margins of safety provided by current TS requirements and limits remain unchanged, as the TS will continue to require operation within the core limits that are based on NRC-approved reload design methodologies. The proposed change does not result in changes to the core operating limits. Appropriate measures exist to control the values of these cycle-specific limits, and appropriate actions will continue to be specified and taken when limits are violated. Such actions remain unchanged.

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

NRC Branch Chief: Robert J. Pascarelli.

STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 18, 2010.

Description of amendment request: The proposed amendments would reduce system/equipment diversity in isolation of low-pressure residual heat removal (RHR) system from high-pressure reactor coolant system (RCS). The change will allow similarly qualified pressure transmitters to be used in more than one RHR train as necessary regardless of manufacturer of the transmitters.

The valves separating the RHR from the RCS are to have independent and diverse interlocks to prevent both from opening unless the RCS pressure is below that of the RHR in compliance with the Nuclear Regulatory Commission's Technical Position ICSB-3, "Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System." Consequently, the change would result in more than minimal increase in the likelihood of a malfunction of systems, structures, or components important to safety as previously evaluated in the plants' Updated Final Safety Analysis Report.

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 consequences of an accident previously evaluated?

   Response: No.

   The proposed change revising the justification for diversity associated with the RHR isolation valves will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The proposed changes will not revise the operability requirements (e.g., leakage limits) for the RHR system. The design-basis accidents will remain the same postulated events described in the STP Unit 1 and Unit 2 Updated Final Safety Analysis Report[,] and the consequences of the design-basis accidents will remain the same.

   Therefore, the proposed changes will not 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 changes will not alter the plant configuration or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. The response of the plant and the operators following an accident will not be different. In addition, the proposed changes do not introduce any new failure modes. In the event the RHR system is overpressurized by the RCS, all leakages originating from RHR components will be detected by the Reactor Coolant Pressure Boundary Leakage Detection System as discussed in the STP UFSAR[Updated Final Safety Analysis Report].

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

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

   Response: No.

   The proposed change to revise the rationale for diversity associated with RHR system isolation valve operation will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The operability requirements for the isolation valves have not been changed, and the RHR system will continue to function as assumed in the safety analysis. In addition, the proposed changes will not adversely affect equipment design or operation, and there are no changes being made to required safety limits or safety system settings that would adversely affect plant safety.

   Therefore, the proposed changes will not result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: Michael T. Markley.

STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 18, 2010.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS) 6.8.3.l, "Containment Post-Tensioning System Surveillance Program." TS 6.8.3.l states that the containment post-tensioning system surveillance program shall be in accordance with American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IML, 1992 Edition with 1992 Addenda, as supplemented by 10 CFR 50.55a(b)(2)(viii). The current inspection interval of South Texas Project (STP), Units 1 and 2 ends in September 2010. The proposed amendments will provide for updating the surveillance program consistent with the updated edition of the ASME Code, Section XI as required by 10 CFR 50.55a.

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 Technical Specification change removes the specific edition of the ASME [C]ode to be applied. Inspection practices will continue to be consistent with the approved ASME [C]ode edition. The proposed change is consistent with NUREG-1481 [guidance].

   Therefore, the proposed changes will not 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 changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. The response of the plant and the operators following an accident will not be different. In addition, the proposed change does not introduce any new failure modes.

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

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

   Response: No.

   The proposed Technical Specification change removes the specific edition of the ASME [C]ode to be applied. Inspection practices will continue to be consistent with the approved ASME [C]ode edition. The change is consistent with NUREG-1481 guidance.

   Therefore, the proposed changes will not result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: Michael T. Markley.

STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: June 28, 2010.

Description of amendment requests: The proposed amendments request correction of an oversight in previous amendments (Amendment No. 185 to Facility Operating License No. NPF-

76 and Amendment No. 172 to Facility Operating License No. NPF-80) that revised the Technical Specifications (TSs) regarding control room envelope (CRE) habitability in accordance with TS Task Force (TSTF) Traveler No. 448, Revision 3. In its application for those previous amendments, STP Nuclear Operating Company (STPNOC) did not specify what shutdown actions would be taken if required actions for an inoperable CRE boundary were not met. This was inconsistent with TSTF-448. The proposed amendments would correct this oversight. STPNOC also requested to add a note to the required actions for inoperable CRE boundary to clarify that the boundary is not a required system, subsystem, train, component, or device that depends on diesel generator as a source of emergency power. This change would clarify the application of TS action 3.8.1.1, "AC Sources, DC Sources, and Other Power Distribution," when the CRE is inoperable.

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

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

   Response: No.

   The proposed change to add the shutdown actions to TS ACTION 3.7.7.d is consistent with Nuclear Regulatory Commission (NRC) noticed Industry/Technical Specification Task Force (TSTF) Standard Technical Specification (STS) change TSTF-448 Revision 3, which has been approved by an NRC safety evaluation.

   The proposed change to add a note to the required action for an inoperable control room envelope boundary does not change the design function of the Control Room Makeup and Cleanup Filtration Systems or the design function of the A.C. Sources, D.C. Sources, and Onsite Power Systems or how these systems operate. The change only clarifies that the Control Room Envelope boundary is not a required system, subsystem, train, component, or device that depends on a diesel generator as a source of emergency power.

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 to add the shutdown actions to TS ACTION 3.7.7.d is consistent with Nuclear Regulatory Commission (NRC) noticed Industry/Technical Specification Task Force (TSTF) Standard Technical Specification (STS) change TSTF-448 Revision 3, which has been approved by an NRC safety evaluation.

   The proposed change to add a note to the required action for an inoperable control room envelope boundary does not change the design of the Control Room Makeup and Cleanup Filtration Systems or the design function of the A.C. Sources, D.C. Sources, and Onsite Power Systems. The change only clarifies that the Control Room Envelope boundary is not a required system, subsystem, train, component, or device that depends on a diesel generator as a source of emergency power.

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

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

   Response: No.

   The proposed change to add the shutdown actions to TS ACTION 3.7.7.d is consistent with Nuclear Regulatory Commission (NRC) noticed Industry/Technical Specification Task Force (TSTF) Standard Technical Specification (STS) change TSTF-448 Revision 3, which has been approved by an NRC safety evaluation.

   The proposed change to add a note to the required action for an inoperable control room envelope boundary does not change any safety margins associated with operation of the Control Room Makeup and Cleanup Filtration Systems or any safety margins associated with the A.C. Sources, D.C. Sources, and Onsite Power Systems. The change only clarifies that the Control Room Envelope boundary is not a required system, subsystem, train, component, or device that depends on a diesel generator as a source of emergency power.

   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 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: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: Michael T. Markley.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING 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.

Notice of Consideration of Issuance of Amendment to Facility Operating License, 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 Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, 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 Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr.resource@nrc.gov.

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

Date of application of amendments: August 31, 2009, as supplemented April 14, 2010.

Brief description of amendments: The amendments revised the Technical Specifications to allow one of the two required 230 kV switchyard 125 Vdc power sources (batteries) to be inoperable for up to 10 days for the purpose of replacing an entire battery bank and performing the required testing.

Date of Issuance: August 30, 2010.

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

Amendment Nos.: 370, 372, 371.

Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: Amendments revised the licenses and the technical specifications.

Date of initial notice in *Federal Register*: March 9, 2010 (75 FR 10828).

The supplement dated April 14, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 30, 2010.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 10, 2009, as supplemented by letters dated December 8, 2009, and April 22, June 16, and August 17, 2010, and by emails dated June 29, July 12, and July 28, 2010.

Brief description of amendment: The amendment revised the TSs for the RBS to support operation with 24-month fuel cycles. By letter dated June 16, 2010, Entergy withdrew its proposed changes to TS 3.3.8 regarding the change to the degraded voltage instrumentation allowable values as indicated on Table 3.3.8.1-1 and to extend the Surveillance Requirement (SR) 3.3.8.1.3 and SR 3.3.8.1.4 from 18 to 24 months. By letter dated August 17, 2010, Entergy withdrew the request for not revising SR 3.3.8.1.4 and requested that this SR be extended as originally requested.

Date of issuance: August 31, 2010.

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

Amendment No.: 168.

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

Date of initial notice in *Federal Register*: October 20, 2009 (74 FR 53776).

The supplements dated December 8, 2009, April 22, June 16, and August 17, 2010, and emails dated June 29, July 12, and July 28, 2010, 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 August 31, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 25, 2009 supplemented by letter dated May 3, 2010.

Brief description of amendment: The amendment modifies technical specification 5.5.14, "Containment Leakage Rate Testing Program," to allow a one-time extension to the 10-year frequency for the next 10 CFR Part 50 Appendix J, Option B, Type A, containment integrity leakage test (ILRT) or Type A test at Palisades Nuclear Plant. This amendment permits the existing ILRT frequency to be extended from 10 years (120 months) to approximately 11.25 years (135 months). This amendment also prevents the necessity of performing a Type A test six months prior to the 10th anniversary of the completion of the last Type A test, which was completed on May 3, 2001.

Date of issuance: August 23, 2010.

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

Amendment No.: 240.

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

Date of initial notice in *Federal Register*: October 20, 2009 (74 FR 53777).

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 23, 2008, as supplemented by letters dated September 28, and November 18, 2009, March 29, and August 3, 2010.

Brief description of amendments: The amendments revise the Technical Specifications to support the application of alternative source term methodology with respect to the loss-of-coolant accident and the fuel-handling accident.

Date of issuance: September 6, 2010.

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

Amendment Nos.: 197, 184.

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications and License.

Date of initial notice in *Federal Register*: April 7, 2009 (74 FR 15771).

The September 28, and November 18, 2009, March 29, and August 3, 2010 supplements 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 amendments is contained in a Safety Evaluation dated September 6, 2010.

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 (PBAPS), Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: August 31, 2009.

Brief description of amendments: The amendments modify the PBAPS Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." Additionally, the change adds a new program, the Surveillance Frequency Control Program, to TS Section 5, Administrative Controls. The changes are based on NRC-approved Industry Technical Specifications Task Force (TSTF) Traveler 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force Initiative 5b," with optional changes and variations as described in Attachment 1, Section 2.2 of the licensee's submittal dated August 31, 2009.

Date of issuance: August 27, 2010.

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

Amendment Nos.: 278 and 281.

Renewed Facility Operating License Nos. DPR-44 and DPR-56: Amendments revised the License and Technical Specifications.

Date of initial notice in *Federal Register*: May 5, 2010 (75 FR 23815).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 27, 2010.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 16, 2010, as supplemented on July 9, 2010.

Description of amendment request: This amendment revises the Seabrook Technical Specifications requirement that the Operations Manager shall have held a senior reactor operator license for the Seabrook Station prior to assuming the Operations Manager position. Specifically, the proposed change now requires the Operations Manager to meet one of the following: (1) hold a senior operator license; (2) have held a senior operator license for a similar unit; or (3) have been certified for equivalent senior operator knowledge.

Date of issuance: September 2, 2010.

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

Amendment No.: 124.

Facility Operating License No. NPF-86: The amendment revised the TS and the License.

Date of initial notice in *Federal Register*: May 4, 2010 (75 FR 23816).

The supplemental letter dated July 9, 2010, 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 September 2, 2010.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: March 29, 2010, as supplemented on June 25, and August 18, 2010.

Brief description of amendments: The amendment revises the Technical Specifications (TSs) to allow a one-time replacement of the 2C 125-volt direct current battery while Salem Unit No. 2 is at power.

Date of issuance: September 1, 2010.

Effective date: As of the date of issuance, to be implemented within 30 days.

Amendment No.: 280.

Facility Operating License No. DPR-75: The amendment revised the TSs and the License.

Date of initial notice in *Federal Register*: June 1, 2010 (75 FR 30446).

The letters dated June 25, and August 18, 2010, 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 amendment is contained in a Safety Evaluation dated September 1, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 10th day of September 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation