7590-01-P

NUCLEAR REGULATORY COMMISSION

BIWEEKLY NOTICE

[NRC-2010-0169]

APPLICATIONS AND AMENDMENTS TO FACILITY OPERATING LICENSES INVOLVING NO SIGNIFICANT HAZARDS CONSIDERATIONS

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 April 8, 2010 to April 21, 2010. The last biweekly notice was published on April 20, 2010 (75 FR 20627).

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.

Calvert Cliffs Nuclear Power Plant, LLC, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: November 23, 2009.

Description of amendments request: The amendment would modify the licensing basis and the Technical Specifications by allowing for the transition from Westinghouse Turbo fuel to AREVA Advanced CE-14 High Thermal Performance (HTP) fuel in the Calvert Cliffs reactors. The licensee plans to refuel and operate with AREVA fuel beginning with the refueling outage in 2011 for Unit No. 2 and 2012 for Unit No. 1. The transition is planned to occur over three refueling cycles on each unit.

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?

   No.

   The reactor fuel and the analyses associated with it are not accident initiators. The response of the fuel to an accident is analyzed using conservative techniques and the results are compared to approved acceptance criteria. These evaluation results will show that the fuel response to an accident is within approved acceptance criteria for both cores loaded with the new AREVA Advanced CE-14 HTP fuel and cores loaded with both AREVA and Westinghouse Turbo fuel. Therefore, the change in fuel design does not affect accident or transient initiation or consequences.

   The proposed change to the Safety Limit Technical Specification (2.1.1.2) does not require any physical change to any plant system, structure, or component. The change to establish the peak fuel centerline temperature as the safety limit is consistent with the Standard Review Plan (SRP) for ensuring that the fuel design limits are met. Operations and analysis will continue to be in compliance with Nuclear Regulatory Commission

(NRC) regulations. The peak fuel centerline temperature is the basis for protecting the fuel and is consistent with the analogous wording for other pressurized water reactor (PWR) plants. Providing the peak fuel centerline melt temperature as the safety limit does not impact the initiation or the mitigation of an accident.

The proposed change to remove the total planar radial peaking factor (F^T_{XY}, Technical Specification 3.2.2) is based on a methodology change. During and after the transition to AREVA Advanced CE-14 HTP fuel, the core analyses are performed using AREVA methodologies. These methodologies do not use the total planar radial peaking factor (F^T_{XY}) as an initial value in the accident analyses. The linear heat rate algorithm limits are provided by the total integrated radial peaking factor, azimuthal power tilt, and axial shape index. The linear heat rate is evaluated in accordance with NRC-approved methodology and meets acceptance criteria. The total planar radial peaking factor is not an accident initiator and does not play a role in accident mitigation. A number of other changes are also made to remove references to Technical Specification 3.2.2 throughout the Technical Specifications.

Topical reports have been reviewed and approved by the NRC for use in determining core operating limits. The core operating limits to be developed using the new methodologies will be established in accordance with the applicable limitations as documented in the appropriate NRC Safety Evaluation reports. The proposed change to add and remove various topical reports to Technical Specification 5.6.5 enables the use of appropriate methodologies to re-analyze certain events. The proposed methodologies will ensure that the plant continues to meet applicable design criteria and safety analysis acceptance criteria.

The proposed change to the list of NRC-approved methodologies listed in Technical Specification 5.6.5 is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. The proposed change will update the listing of NRC-approved methodologies to remove methods no longer used and add new methods consistent with the transition to AREVA Advanced CE-14 HTP fuel. Changes to the calculated core operating limits may only be made using NRC-approved methods, must be consistent with all applicable safety analysis limits and are controlled by the 10 CFR 50.59 process. The list of methodologies in the Technical Specifications does not impact either the initiation of an accident or the mitigation of its consequences.

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 type of accident from any accident previously evaluated?

   No.

   Use of AREVA Advanced CE-14 HTP fuel in the Calvert Cliffs reactor cores is consistent with the current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety systems, structures, or components, or their roles in accident prevention or mitigation. The operational characteristics of AREVA Advanced CE-14 HTP fuel are bounded by the safety analyses. The AREVA Advanced CE-14 HTP fuel design performs within fuel design limits and does not create the possibility of a new or different type of accident.

The proposed change to the Safety Limit Technical Specification (2.1.1.2) does not require any physical change to any plant system, structure, or component, nor does it require any change in safety analysis methods or results. The existing analyses remain unchanged and do not affect any accident initiators that would create a new accident.

The proposed change to remove the total planar radial peaking factor (F^T_{XY}, Technical Specification 3.2.2) is based on a change in analytical methods needed to support the physical fuel change. These methodologies do not use the total planar radial peaking factor (F^T_{XY}) as an initial value in the accident analysis. The total planar radial peaking factor does not play a role in accident mitigation and cannot create the possibility of a new or different kind of accident. A number of other changes are made to remove references to Technical Specification 3.2.2 throughout the Technical Specifications.

The proposed change to the list of topical reports used to determine the core operating limits is administrative in nature and has no impact on any plant configuration or on system performance. It updates the list of NRC-approved topical reports used to develop the core operating limits. There is no change to the parameters within which the plant is normally operated. The possibility of a new or different accident is not created.

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

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

No.

Use of AREVA Advanced CE-14 HTP fuel is consistent with the current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety systems, structures, or components, or their roles in accident prevention or mitigation. The operational characteristics of AREVA Advanced CE-14 HTP fuel are bounded by the safety analyses. The AREVA Advanced CE-14 HTP fuel design performs within fuel design limits. The proposed changes do not result in exceeding design basis limits. Therefore, all licensed safety margins are maintained.

The proposed change to the Safety Limit Technical Specification (2.1.1.2) does not require any physical change to any plant system, structure, or component, nor does it require any change in safety analysis methods or results. Therefore, by changing the safety limit from peak linear heat rate to peak fuel centerline temperature, the margin as established in the current licensing basis remains unchanged.

The proposed change to remove the total planar radial peaking factor (F^T_{XY},Technical Specification 3.2.2) is based on a methodology change. The linear heat rate algorithm limits are provided by the total integrated radial peaking factor, azimuthal power tilt, and axial shape index. The linear heat rate is evaluated in accordance with NRC-approved methodology and meets acceptance criteria. Therefore, the margin as established for the linear heat rate remains unchanged. A number of other changes are made to remove references to Technical Specification 3.2.2 throughout the Technical Specifications.

The proposed change to the list of topical reports does not amend the cycle specific parameters presently required by the Technical Specifications. The individual Technical Specifications continue to require operation of the plant within the bounds of the limits specified in the COLR [Core Operating Limits Report]. The proposed change to the list of analytical methods referenced in the COLR is administrative in nature and does not impact 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 amendments request involves no significant hazards consideration.

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

NRC Branch Chief: Nancy L. Salgado

Entergy Nuclear Operations, Inc., Docket No. 50-293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: March 15, 2010.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) to address the increased setpoints and setpoint tolerances for Safety Relief Valves (SRVs) and Spring Safety Valves (SSVs) and changes related to the replacement of four Target Rock two-stage SRVs with more reliable three-stage SRVs and two existing Dresser 3.749 inch throat diameter SSVs with Dresser 4.956 inch diameter SSVs.

<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 increases the allowable as-found SRV and SSV setpoint tolerance, determined by test after the valves have been removed from service, from ±1% to ± 3%. The proposed change also increases the SRV and SSV setpoints. Analysis of these changes demonstrates that reactor pressure will be maintained below the applicable code overpressure limits. The proposed change increases the SSV discharge capacity due its increased throat diameter. The proposed change does not alter the TS requirements for the number of SRVs and SSVs required to be operable, the allowable as-left lift setpoint tolerance, the testing frequency, or the manner in which the valves are operated. Consistent with current TS requirements, the proposed change continues to require that the safety valves be adjusted to within ± 1 % of their nominal lift setpoints following testing. The proposed increase in the SRV and SSV setpoint complies with the ASME Boiler and Pressure Vessel (B&PV) Code (1965 Edition, including January 1966 Addendum) for the pressure vessel, USAS Piping Code Section B31.1 for the steam space piping, and ASME Section III for the reactor coolant system recirculation piping. Since the proposed change does not alter the manner in which the valves are operated, there is no significant impact on the reactor operation.

   The proposed change does not involve a change to the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions. Therefore, these changes will not increase the probability of an accident previously evaluated.

   Since an SSV setpoint increase and setpoint tolerance will increase the SSV safety valve opening pressure and an increase in the SSV throat size will increase the SSV flow capacity, the SSV dynamic loads are expected to increase. Entergy has evaluated the SSV dynamic loads for the associated piping. All piping and structures were found to meet Code requirements.

   Since an SRV setpoint and the setpoint tolerance increase will increase the SRV valve opening pressure, the SRV discharge dynamic loads will increase. Entergy has evaluated the SRV dynamic load increases for the associated piping and torus submerged structures and the evaluation concluded that all piping and structures were found to meet Code requirements.

The proposed revision to the HPCI [high-pressure coolant injection] and RCIC [Reactor Core Isolation Cooling] pump operability determination surveillance follows the format of BWR Standard Technical Specification surveillance, and complies with in-service testing for pump operability determination in accordance with ASME OM Code requirement.

Generic considerations related to the change in setpoints and setpoint tolerance were addressed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," and were reviewed and approved by the NRC in a safety evaluation dated March 8, 1993. General Electric Hitachi Company (GEH) completed plant-specific analyses to assess the impact of increase in SRV and SSV setpoints and increase in the setpoint tolerance from ± 1% to ± 3%. The impact of the increases in the SRV and SSV setpoints and increases in the setpoint tolerances, as addressed in this analyses, included vessel overpressure, Updated Final Safety Analysis Report (UFSAR) Chapter 14 events, ATWS [Anticipated Transient Without Scram], Loss of Coolant Accident (LOCA), containment response and dynamic loads, high pressure systems performance, operating mode and equipment out of service. The proposed change is supported by GEH analysis of events that credit the SRVs and SSVs.

The plant specific evaluations, required by the NRC's safety evaluation and performed to support this proposed change, demonstrate that there is no change to the design core thermal limits and adequate margin to the reactor coolant system pressure limits exists. These analyses also demonstrate that operation of Core Standby Cooling Systems (CSCS) is not adversely affected and the containment response following a LOCA is acceptable. The plant systems associated with these proposed changes are capable of meeting applicable design basis requirements and retain the capability to mitigate the consequences of accidents described in the UFSAR. Therefore, these changes do not involve an increase in the consequences of an accident previously evaluated.

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

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

   Response: No.

   The proposed change increases the allowable as-found lift setpoint tolerance for the Pilgrim SRV and SSV valves. The proposed change to increase the tolerance was developed in accordance with the provisions contained in the NRC safety evaluation for NEDC-31753P. SRVs and SSVs installed in the plant following testing will continue to meet the current tolerance acceptance criteria of ± 1% of the nominal setpoint. The proposed change does not affect the manner in which the overpressure protection system is operated; therefore, there are no new failure mechanisms for the overpressure protection system.

The proposed changes do not change the safety function of the SRVs and SSVs, or HPCI and RCIC systems. There is no alteration to the parameters within which the plant is normally operated. The increase in SRV and SSV setpoints, setpoint tolerance, and increased SSV discharge capacity are not precursors to new or different kind of accidents and do not initiate new or different kind of accidents. The impact of these changes have been analyzed and found to be acceptable within the design limits and plant operating procedures.

As a result, no new failure modes are being introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change modifies the setpoints at which protective actions are initiated, and [...] does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

Establishment of the ± 3% SRV and SSV setpoint tolerance limit does not adversely affect the operation of any safety-related component or equipment. Evaluations performed in accordance with the NRC safety evaluation for NEDC-31753P have concluded that all design limits will continue to be met.

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.

<u>Attorney for licensee</u>: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601.

<u>NRC Branch Chief</u>: Nancy Salgado.

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

Date of amendment request: February 22, 2010.

Description of amendment request: The proposed amendment will modify Technical Specification (TS) 3/4.9.4, "Containment Building Penetrations," to allow alternative means of penetration closure during Core Alterations or irradiated fuel movement while in refueling operations. Additional improvements to the TS are also being proposed, as well as the elimination of TS 3/4.9.9, "Containment Purge Valve Isolation System." The proposed changes are consistent with Revision 3 of NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants."

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.

   TS 3/4.9.4 currently allows containment penetration flow paths to be open during Core Alterations or movement of irradiated fuel within containment under specific administrative controls. The proposed change would allow additional approved methods for ensuring positive penetration closure. The fuel handling accident (FHA) radiological analysis does not take credit for containment isolation or filtration. Therefore, the time required to close any open penetrations does not affect the radiological analysis dose calculations and the proposed change does not involve a significant increase in the consequences of an accident previously evaluated. The administrative controls for containment penetration closure are conservative even though not required by the accident analysis.

   The proposed revision only provides alternate methods of penetration closure and does not alter any plant equipment where the probability of an accident would be increased. The incorporation of purge valve isolation surveillance requirements for assuring purge valve Operability has no affect on the probability or consequences of the analyzed accidents.

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.

   Alternative methods of providing penetration closure do not create accident initiators and do not represent a significant change in the configuration of the plant. The proposed allowance to secure containment penetrations during refueling operations will not adversely affect plant safety functions or equipment operating practices such that a new or different accident could be created.

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

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

   Response: No.

   TS Limiting Condition for Operation (LCO) 3.9.4 closure requirements for containment penetrations ensure that the consequences of a postulated FHA inside containment during Core Alterations or fuel handling activities are minimized. The LCO establishes containment closure requirements, which limit the potential escape paths for fission products by ensuring that there is at least one barrier to the release of radioactive material. The proposed change to allow alternate methods of reaching containment penetration closure during Core Alterations or fuel movement does not affect the expected dose consequences of a FHA since it does not credit containment building closure. The proposed administrative controls provide assurance that prompt closure of the penetration flow paths will be accomplished in the event of a FHA inside containment thus minimizing the transmission of radioactive material from the containment to the outside environment. The incorporation of purge valve isolation surveillance requirements does not reduce any margins 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: Joseph A. Aluise, Associate General Council - Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: February 24, 2010.

Description of amendment request: The proposed amendment deletes Operating License Condition 2.C.14 (Fuel Movement in the Fuel Handling Building) due to electing to comply with Section 50.68, "Criticality accident requirements," of Title 10 of the *Code of Federal Regulations* (10 CFR). The Operating License Condition 2.C.14, "no more than one fuel assembly shall be out of its shipping container or storage location at a given time," was one basis for the exemption from the criticality alarm system requirements of 10 CFR 70.24. The criticality accident requirements can be met either by complying with 10 CFR 70.24 or 10 CFR 50.68 requirements. The 10 CFR 50.68 criteria are now being used; therefore, Operating License Condition 2.C.14 is no longer applicable.

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 amendment deletes Operating License Condition 2.C.14 (Fuel Movement in the Fuel Handling Building) due to electing to comply with 10CFR50.68 requirements.

   The proposed changes will not alter the configuration of the storage racks or their environment. The fuel racks will not be operated outside of their design limits, and no additional loads will be imposed on them. Therefore, these changes will not affect fuel storage rack performance or reliability. No new equipment will be introduced into the plant. The accuracies and response characteristics of existing instrumentation will not be modified. The proposed changes will not require, or result in, a change in safety system operation, and will not affect any system interface with the fuel storage racks. Fuel assembly placement will continue to be controlled in accordance with approved fuel handling procedures. All the requirements of 10CFR50.68 continue to be met which ensures no significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will not affect any barrier that mitigates dose to the public, and will not result in a new release pathway being created. The functions of equipment designed to control the release of radioactive material will not be impacted, and no mitigating actions described or assumed for an accident in the UFSAR [Updated Final Safety Analysis Report] will be altered or prevented. No assumptions previously made in evaluating the consequences of an accident will need to be modified. Onsite dose will not be increased, so the access of plant personnel to vital areas of the plant will not be restricted, and mitigating actions will not be impeded.

Therefore, it is concluded that the proposed changes do not significantly increase either the probability or consequences of any accident previously evaluated.

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

   Response: No.

   The proposed amendment deletes Operating License Condition 2.C.14 (Fuel Movement in the Fuel Handling Building) due to electing to comply with 10 CFR 50.68 requirements.

   10 CFR 50.68(b)(1) provides the requirements to ensure that plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water. By meeting this criteria, the removal of Operating License Condition 2.C.14 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

   Response: No.

> The proposed amendment deletes Operating License Condition 2.C.14 (Fuel Movement in the Fuel Handling Building) due to electing to comply with 10 CFR 50.68 requirements.
>
> 10 CFR 50.68(b)(1) provides similar requirements as that contained in Operating License Condition 2.C.14. The NRC has approved the [Waterford Steam Electric Station, Unit 3] use of 10 CFR 50.68 criteria. By meeting the 10 CFR 50.68(b)(1) requirements, there will not be a significant reduction in a margin of safety.
>
> Therefore, it is 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, 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 Council - Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

Exelon Generation Company, LLC, Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: February 15, 2010.

Description of amendment request: The proposed amendment would relocate selected Surveillance Requirement frequencies from the Clinton Power Station, Unit No. 1 (Clinton) Technical Specifications (TSs) to a licensee-controlled program. This change is based on the NRC-approved Industry Technical Specifications Task Force (TSTF) change TSTF-425, “Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b,” Revision 3, (Agencywide Documents Access and Management System (ADAMS) Accession Package No. ML090850642). Plant-specific deviations from TSTF-425 are proposed to accommodate differences between the Clinton TSs and the model TSs originally used to develop TSTF-425.

The Nuclear Regulatory Commission (NRC) staff issued a Notice of Availability for TSTF-425 in the *Federal Register* on July 6, 2009 (74 FR 31996). The notice included a model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination. In its application dated February 15, 2010 (ADAMS Accession No. ML100470787), the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

Response: No. The proposed change relocates the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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 previously evaluated?

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

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 the margin of safety?

Response: No. The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Exelon will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-01, Rev. 1. The methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177 [An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications].

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

NRC Branch Chief: Stephen J. Campbell.

Exelon Generation Company, LLC, Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: March 3, 2010.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time (CT) for Condition B (i.e., "Two SLC subsystems inoperable") from 8 hours to 72 hours.

Basis for proposed no significant hazards consideration**:** 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 any accident previously evaluated?

   Response: No.

   The proposed amendment revises Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time (CT) for Condition B (i.e., "Two SLC subsystems inoperable.") from eight hours to 72 hours.

   The proposed change is based on a risk-informed evaluation performed in accordance with Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and RG 1.I77, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications."

The proposed amendment modifies an existing CT for a dual-train SLC system inoperability. The condition evaluated, the action requirements, and the associated CT do not impact any initiating conditions for any accident previously evaluated.

The proposed amendment does not increase postulated frequencies or the analyzed consequences of an Anticipated Transient Without Scram (ATWS). Requirements associated with 10 CFR 50.62 will continue to be met. In addition, the proposed amendment does not increase postulated frequencies or the analyzed consequences or a large-break loss-of-coolant accident for which the SLC system will be used for pH control. The extended CT provides additional time to implement actions in response to a dual-train SLC system inoperability, while also minimizing the risk associated with continued operation. 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 previously evaluated?

   Response: No.

   The proposed amendment revises TS 3.1.7 to extend the CT for Condition B from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. This proposed TS amendment does not change the design function of the SLC system and does not affect the system's ability to perform its design function. The SLC system provides a method to bring the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. Required actions and surveillance requirements are sufficient to ensure that the SLC system functions are maintained. No new accident initiators are introduced by this amendment. Therefore, the proposed amendment 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 amendment revises TS 3.1.7 to extend the CT for Condition B from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the setpoints for the actuation of equipment relied upon to respond to an event.

   The proposed amendment does not modify the condition or point at which SLC is initiated, nor does it affect the system’s ability to perform its design function. In

> addition, the proposed change complies with the intent of the defense-in-depth philosophy and the principle that sufficient safety margins are maintained, consistent with RG 1.177 requirements (i.e., Section C, “Regulatory Position,” paragraph 2.2 “Traditional Engineering considerations”).
>
> Based on the above analysis, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

The NRC staff has reviewed the analysis adopted by the licensee 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. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

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 amendment request: August 31, 2009.

Description of amendment request: The proposed amendment would 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 would add 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.

<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. Do the proposed changes involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No.

   The proposed changes relocate the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program [SFCP]. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

   Response: No.

   No new or different accidents result from utilizing 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) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

Response: No.

> [… T]here is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Exelon will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the TS SFCP. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177. 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, and with the changes noted above, 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. J. Bradley Fewell, Associate General Counsel, Exelon Generation Company LLC, 4300 Winfield Road, Warrenville, IL 60555.

<u>NRC Branch Chief</u>: Harold K. Chernoff

<u>FPL Energy Seabrook, LLC Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire</u>

<u>Date of amendment request</u>: March 16, 2010.

<u>Description of amendment request</u>: The proposed changes would revise 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 would require 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. In its

application dated March 16, 2010, the licensee concluded that the no significant hazards consideration (NSHC) determination presented in the notice is applicable to Seabrook Station.

Basis for proposed NSHC determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of NSHC, 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 requested change would only affect the qualification requirements for the Operations Manager Position]. The proposed change does not impact the configuration or function of plant structures, systems, or components (SSCs) or the manner in which SSCs are operated, maintained, modified, tested, or inspected. No actual facility equipment or accident analyses will be affected by the proposed changes. Therefore, this request has no [significant] impact on 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 accident previously evaluated.

   [The requested change would only affect the qualification requirements for the Operations Manager Position]. The proposed change does not alter the plant configuration, require new plant equipment to be installed, alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected. Therefore, this request does no create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a 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 requested change would only affect the qualification requirements for the Operations Manager Position]. No actual plant equipment or accident analyses will be affected by the proposed changes. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety system settings, and will not relax the bases for any limiting conditions for operation. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis. The proposed change does not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. Therefore, these 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, 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.

Attorney for licensee: M.S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420.

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: November 24, 2009.

Description of amendment request: The proposed amendments would make changes to Technical Specification (TS) Section 4.2.1, Fuel Assemblies, and TS Section 5.6.5, Core Operating Limit Report, by revising the TS to allow the use of Optimized ZIRLO™ fuel rod cladding material.

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.

   Westinghouse Electric Company, LLC (Westinghouse) topical report WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A "Optimized ZIRLO™", July 2006, provides the details and results of material testing of Optimized ZIRLO™ compared to standard ZIRLO™ as well as the material properties to be used in various models and methodologies when analyzing Optimized ZIRLO™. The Nuclear Regulatory Commission (NRC) has allowed use of Optimized ZIRLO™ fuel cladding material in Westinghouse fueled reactors provided that licensees ensure compliance with the conditions and limitations set forth in the NRC Safety Evaluation (SE) for the topical report. By satisfying the conditions and limitations of the NRC SE through completed actions and its approved reload safety evaluation process, the licensee ensures that the effects of Optimized ZIRLO™ on PINGP core performance are evaluated and that the probability or consequences of previously-evaluated accidents are not increased.

Therefore, the proposed change of adding a cladding material does not result in an increase to 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.

   Material properties of this fuel design have been evaluated in Westinghouse topical report WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A "Optimized ZIRLO™" July 2006. That report provides the details and results of material testing of Optimized ZIRLO™ compared to standard ZIRLO™ as well as the material properties to be used in various models and methodologies when analyzing Optimized ZIRLO™. Neither that topical report nor the associated NRC SE identifies the possibility of a new or different kind of accident resulting from this change for generic application in Westinghouse reactors. As demonstrated in that topical report and stated in the NRC SE, there is reasonable assurance that under both normal and accident conditions, the Optimized ZIRLO™ fuel cladding will be able to safely operate and comply with NRC regulations. By satisfying the conditions and limitations of the NRC SE by virtue of its completed actions and its approved reload safety evaluation process, the licensee ensures that the effects of Optimized ZIRLO™ are evaluated and will not create the possibility of a new or different kind of accident. Assurance that the possibility of new or different type of accidents will not be created on a site-specific basis is inherent to the reload safety evaluation process approved for use at the PINGP. Site specific evaluation of the PINGP core designs with Optimized ZIRLO™ will be performed programmatically and necessarily by the approved reload safety evaluation process.

   Therefore, the proposed change of adding a cladding material 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 cladding material used in the fuel rods is designed and tested to prevent excessive fuel temperatures, excessive internal rod gas pressure due to fission gas releases, and excessive cladding stresses and strains. Optimized ZIRLO™

> was developed to meet these needs and provides a reduced corrosion rate while maintaining the benefits of mechanical strength and resistance to accelerated corrosion from abnormal chemistry conditions. Westinghouse topical report WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A "Optimized ZIRLO™", July 2006, provides the details and results of material testing of Optimized ZIRLO™ compared to standard ZIRLO™ as well as the material properties to be used in various models and methodologies when analyzing Optimized ZIRLO™. The NRC has allowed use of Optimized ZIRLO™ fuel cladding material detailed within this topical report as detailed within their SE. Therefore, the change in material does not result in 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.

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: January 27, 2010.

Description of amendment request: The proposed amendments would make changes to the Technical Specifications (TS) to revise TS 3.8.3, “Diesel Fuel Oil”. The amendments would revise the diesel fuel oil (DFO) storage volumes applicable to Unit 1 in TS 3.8.3 Condition statements A and D, and increase the Unit 1 DFO supply required by surveillance requirement 3.8.3.1. The amendments would clarify wording in TS 3.8.3 Condition B statement which applies to both units.

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

   This license amendment request proposes to increase the emergency diesel generator fuel oil storage volumes specified in the Technical Specification Condition statements and Surveillance Requirements. Also a word was added to a Condition statement to clarify its meaning.

   The emergency diesel generators and their supporting diesel fuel oil storage systems are not accident initiators and therefore the proposed fuel oil storage volume increases do not involve an increase in the probability of an accident.

   The proposed increased diesel fuel oil storage volumes provide sufficient volumes to maintain the current licensing basis for emergency diesel generator operation. Thus the proposed fuel oil storage volume increases do not involve a significant increase in the consequences of an accident.

   The proposed Technical Specification Condition statement wording clarification is administrative and thus does not involve an increase in the probability of an accident or an increase in the consequences of an accident.

   Therefore, the proposed Technical Specification 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

   This license amendment request proposes to increase the emergency diesel generator fuel oil storage volumes specified in the Technical Specification Condition statements and Surveillance Requirements. Also a word was added to a Condition statement to clarify its meaning.

   The proposed Technical Specification changes which increase emergency diesel generator fuel oil storage volumes do not change any system operations or maintenance activities. The changes do not involve physical alteration of the plant, that is, no new or different type of equipment will be installed. The changes do not alter assumptions made in the safety analyses but ensures that the diesel generators operate as assumed in the accident analyses. These changes do not create new failure modes or mechanisms which are not identifiable during testing and no new accident precursors are generated.

The proposed Technical Specification Condition statement wording clarification is administrative and thus does not create the possibility of a new or different kind of accident.

Therefore, the proposed Technical Specification changes do 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

This license amendment request proposes to increase the emergency diesel generator fuel oil storage volumes specified in the Technical Specification Condition statements and Surveillance Requirements. Also a word was added to a Condition statement to clarify its meaning.

Since this license amendment proposes Technical Specification changes which increase the required fuel oil storage volumes, margins of safety are increased and thus no margin of safety is reduced as part of this change.

The proposed Technical Specification Condition statement wording clarification is administrative and thus does not involve a significant reduction in a margin of safety.

Therefore, the proposed Technical Specification 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 requests involve no significant hazards consideration.

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

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

Southern Nuclear Operating Company, Inc., Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: February 2, 2010.

Description of amendment request: The proposed amendments would revise the verification requirements for the Reactor Trip System Instrumentation. Specifically, the amendment proposes the addition to Table 3.3.1-1 of a response time measurement for the verification of the Power Range Neutron High Positive Rate Trip (PFRT) function as recommended by Westinghouse Nuclear Safety Advisory Letter (NSAL-09-01) "Rod Withdrawal at Power Analysis for Reactor Coolant System Overpressure."

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

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

   Response: No.

   The proposed change to Vogtle Electric Generating Plant (VEGP) Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," Table 3.3.1-1, "Reactor Trip System Instrumentation" does not significantly increase the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The overall protection system performance will remain within the bounds of the accident analysis since there are no hardware changes. The design of the Reactor Trip System (RTS) instrumentation, specifically the positive range neutron flux high positive rate trip (PFRT) function, will be unaffected. The reactor protection system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

   The proposed change adds an additional surveillance requirement to assure that the PFRT is verified to be consistent with the safety analysis and licensing basis. In this specific case, a response time verification requirement will be added to the PFRT function.

   The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed change will not alter any assumptions nor change any mitigation actions in the radiological consequences evaluations in the UFSAR.

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter nor prevent the ability of SSCs from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change is consistent with the safety analyses assumptions and resultant consequences. The RCS overpressure limit listed in Specification 2.1.2 of the VEGP Technical Specifications (i.e., 2735 psig) is not violated.

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

   Response: No.

   There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. This change will not affect the normal method of plant operation nor change any operating parameters.

   No performance requirements will be affected; however, the proposed change adds an additional surveillance requirement. The additional surveillance requirement is consistent with assumptions made in the safety analyses and licensing basis.

   No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change.

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

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

   Response: No.

   The proposed change does not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Limits. There will be no effect on the manner in which Safety Limits or Limiting Conditions of Operations are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

> This change is consistent with the assumptions made in the safety analyses. The addition of a surveillance requirement increases the margin of safety by assuring that the associated safety analysis assumption on the PFRT response time is verified.
>
> Therefore, the proposed change does not involve a significant reduction in the margin of safety.
>
> Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standard set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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

PREVIOUSLY PUBLISHED NOTICES OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the *Federal Register* on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: March 31, 2010.

Brief description of amendment request: The proposed amendment would add new license condition 2.C(4) stating that performance of Technical Specification surveillance requirement 3.1.4.3, which verifies control rod freedom of movement, is not required for control rod drive 22 during cycle 21 until the next entry into Mode 3 in a maintenance or refueling outage, whichever is earlier.

Date of publication of individual notice in FEDERAL REGISTER: April 14, 2010 (75 FR 19428).

Expiration date of individual notice: June 13, 2010.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: March 29, 2010, as supplemented by letter dated March 29, 2010.

Brief description of amendment request: The proposed amendment would revise the Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," regarding function 6.g in TS Table 3.3.2-1. Function 6.g provides an auxiliary feedwater (AFW) start signal that is provided to the motor-driven AFW pumps in the event of a trip of both turbine-driven main feedwater pumps. The changes would revise Condition J for ESFAS instrumentation function 6.g to read, "One or more Main Feedwater Pumps trip channel(s) inoperable." The licensee will make corresponding changes to Required Action J.1 and the Note above Required Actions J.1 and J.2 for consistency with the revised Condition.

Date of publication of individual notice in FEDERAL REGISTER: April 14, 2010 (75 FR 19431).

Expiration date of individual notice: April 28, 2010, for public comments; June 14, 2010, for hearing requests.

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.

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

Date of application for amendment: November 23, 2009, as supplemented by letter dated February 5, 2010.

Brief description of amendment: The amendment modified the Technical Specification (TS) 5.5.7, "Inservice Testing Program," by replacing the references from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to the current Code of Record, the ASME Operation and Maintenance Nuclear Power Plants Code (ASME OM Code), the Code of Record for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Inservice Testing

(IST) Program. This is an administrative amendment to maintain the TS current with the NRC accepted Code of Record for JAFNPP IST Program.

Date of issuance: April 12, 2010.

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

Amendment No.: 296.

Renewed Facility Operating License No. DPR-59: The amendment revised the License and the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: January 26, 2010 (75 FR 4117).

The February 5, 2010, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC 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 April 12, 2010.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50-454 and STN 50-455, Byron Station (Byron), Unit Nos. 1 and 2, Ogle County, Illinois

Date of application for amendment: September 24, 2009, as supplemented by letters dated November 13, 2009; January 19, 2010; March 1, 2010; March 9, 2010 (two letters); and March 19, 2010.

Brief description of amendment: The amendments adds a new Completion Time (CT) of

144 hours to restore a unit-specific essential service water train to operable status associated with the Limiting Condition for Operation for Technical Specification (TS) 3.7.8, "Essential Service Water (SX) System." The new CT will be used for maintenance during the Byron, Unit No. 2, spring 2010, refueling outage. The licensee requested the new CT to replace two of the four SX pump suction isolation valves without having to shutdown Byron, Unit No. 1; maintenance history has shown that replacement of the SX pump suction isolation valves cannot be assured within the existing 72 hour CT window.

Date of issuance: April 9, 2010.

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

Amendment Nos.: Unit No. 1 - 168; Unit No. 2 - 168.

Facility Operating License Nos. NPF-37 and NPF-66: The amendments revise the TSs and Licenses.

Date of initial notice in FEDERAL REGISTER: December 1, 2009 (74 FR 62835).

The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 9, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 18, 2009.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.7, “Inservice Testing Program,” by incorporating TS Task Force Traveler (TSTF)-479, “Changes to Reflect Revision of 10 CFR 50.55a,” and TSTF-497, “Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less.” Specifically, the amendments (1) replace references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI with the ASME Code for Operation and Maintenance of Nuclear Power Plants for inservice testing activities, and (2) applies the extension allowance of SR 3.0.2 to other normal and accelerated inservice testing frequencies of 2 years or less that were not included in the frequencies listed in TS 5.5.7.a.

Date of issuance: April 8, 2010.

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

Amendment No.: 110.

Renewed Facility Operating License No. DPR-18: Amendment revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: November 3, 2009 (74 FR 56887).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 8, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 22nd day of April 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Nelson, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation