[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2010-0393]

BIWEEKLY NOTICE

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 December 2, 2010, to December 15, 2010. The last biweekly notice was published on December 14, 2010 (75 FR 77906).

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 O1 F21, 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 O1-F21, 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*.

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

<u>Date of amendment request</u>: July 22, 2010.

<u>Description of amendment request</u>: The amendments would revise an element of the methodology used in evaluating the radiological consequences of design basis steam generator tube rupture (SGTR) accidents. Specifically, the changes will revise the Palo Verde Nuclear Generating Station (PVNGS) Updated Final Safety Analysis Report (UFSAR), Section 15.6.6, "Steam Generator Tube Rupture," to reflect a lower iodine spiking factor assumed for the coincident event Generated Iodine Spike (GIS) and the resulting reduction in the radiological consequences provided in UFSAR Table 15.6.3-5, "Radiological Consequences for the Limiting SGTRLOPSF [Steam Generator Tube Rupture with Loss of Offsite Power and Single Failure] Event.

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

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

   Response: No.

   The proposed amendment changes an element of the methodology used in evaluating the radiological consequences of design basis SGTR accidents. This change will revise the iodine spiking factor used for a GIS from a value of 500 to a value of 335. The proposed change in the methodology element does not involve any design or physical changes to the facility or any component of that facility. The proposed change creates no new failure modes or initiating occurrences that could result in a design basis transient or accident evaluated in the Palo Verde Nuclear Generating Station (PVNGS) Updated Final Safety Analysis Report (UFSAR). Therefore the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change in the methodology element does change the design basis analyses results for PVNGS. However, the results remain bounded by the previous analyzed values and remain within the acceptance criteria for PVNGS of 100% of the 10 CFR [Part] 100 maximum thyroid dose limit of 300 rem [roentgen equivalent man].

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

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 changes an element of the methodology used in evaluating the radiological consequences of design basis SGTR accidents. This change will revise the iodine spiking factor used for a GIS from a value of 500 to a value of 335. The proposed change in the methodology element does not involve any design or physical changes to the facility or any component of that facility. The proposed change in the methodology element does change the design basis analyses results for PVNGS; however, these results remain bounded by the previous analyzed values and remain within the acceptance criteria for PVNGS of 100% of the 10 CFR [Part] 100 maximum thyroid dose limit of 300 rem.

   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 amendment changes an element of the methodology used in evaluating the radiological consequences of design basis SGTR accidents. This change will revise the iodine spiking factor used for a GIS from a value of 500 to a value of 335. The proposed change in the methodology element does not involve any design or physical changes to the facility or any component of that facility. The proposed methodology element change for a postulated SGTR, with a coincident loss of offsite power, GIS, and a failed open atmospheric dump valve (ADV), results in lower maximum dose consequences at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) [than] previously analyzed for this event combination. The methodology element change results in the 2-hour maximum thyroid dose value of 182 rem at the EAB being reduced to 124 rem. In addition, the 8-hour maximum thyroid dose of 125 rem at the LPZ, would be reduced to 84 rem.

> Previously for PVNGS, the GIS 8-hour maximum thyroid dose was bounding at the LPZ and the pre-Accident Iodine Spike (PIS) 2-hour maximum thyroid dose was bounding at the EAB. The methodology element change reduces the GIS calculated dose at both the EAB and LPZ for SGTR events, but it does not affect the PIS dose values. Since the GIS calculated dose at the LPZ drops below the PIS 8-hour LPZ maximum thyroid dose (91 rem), the PIS 8-hour LPZ dose will become bounding for PVNGS. The PIS 2-hour EAB maximum thyroid dose (294 rem), remains the bounding dose at the EAB.
>
> The revised dose consequences remain bounded by the previous analyzed values and remain within the 10 CFR Part 100 guideline values which are the acceptance criteria for PVNGS Units 1, 2, and 3. In addition, the proposed change has no effect on previously reported dose consequences for control room personnel following any postulated SGTR event.
>
> 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 that review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: October 6, 2010.

Description of amendment request: The proposed change will revise the note in Surveillance Requirement (SR) 3.5.4.1 in the Refueling Water Storage Tank (RWST) Technical Specification (TS). Specifically, the proposed change will not require monitoring of the RWST temperature every 24 hours when the RWST heating steam supply isolation valves are locked closed.

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?

   No. The proposed change revises the existing Indian Point [Nuclear Generating Unit No.] 3 [(IP3)] Refueling Water Storage Tank (RWST) Technical Specification (TS) Surveillance Requirement (SR) 3.5.4.1 to revise the note that eliminates the requirement to perform SR 3.5.4.1 when ambient air temperatures are within the operating limits of the RWST. The revision to the note adds a requirement that the steam heating supply isolation valves be locked closed when not performing the surveillance. The additional requirement does not increase the probability of an accident occurring since it is not an accident initiator and does not increase the consequences of an accident since it is providing additional assurance that the RWST is within the temperature limits assumed for accident analyses. The change increases observation of the RWST temperature when the steam supply isolation valves are not locked closed and does not otherwise affect [***] the performance capability of the structures, systems, and components relied upon to mitigate the consequences of postulated 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 change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed change revises the note that eliminates the requirement to perform SR 3.5.4.1 when ambient air temperatures are within the operating limits of the RWST. The revision adds the additional requirement of locking closed the steam supply isolation valves. The proposed change does not involve installation of new equipment or modification of existing equipment, so that no new equipment failure modes are introduced. Also, the proposed change does not result in a change to the way that the equipment or facility is operated so that no new accident initiators are 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 change involve a significant reduction in a margin of safety?

   No. The proposed change revises the note that eliminates the requirement to perform SR 3.5.4.1 when ambient air temperatures are within the operating limits

> of the RWST. The revision adds the additional requirement of locking closed the steam supply isolation valves. The change does not reduce margin since it increases the temperature surveillance frequency for the RWST to provide further assurance that the required water temperature is maintained at all times.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Nancy L. Salgado.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: November 8, 2010.

Description of amendment request: The proposed license amendment request will make changes related to the final resolution of an unresolved issue associated with Technical Specification (TS) Amendment No. 181 dated February 25, 2009. This issue was resolved with the approval of Revision 4 of Technical Specification Task Force (TSTF) Change Traveler TSTF-493, “Clarify Application of Setpoint Methodology for LSSS [Limiting Safety System Setting] Functions,” which included the instrument function (i.e., Condensate Storage Tank (CST) Level-Low) that was the subject of Amendment No. 181. Specifically, the proposed change will add the appropriate notes as specified in TSTF-493 to the surveillance requirements associated with TS Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation," Function 3.d, Condensate Storage Tank Level - Low, and to TS Table 3.3.5.2-1, "Reactor Core Isolation Cooling System Instrumentation," Function 3, Condensate Storage Tank Level - Low. The supporting TS Bases will also be revised.

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 adds test requirements to the CST Level-Low function to ensure the CST Level-low instruments will function as required. Surveillance tests are not an initiator of any accident previously evaluated. The CST components, for which the additional requirements were added, continue to be operable and capable of performing their intended function.

   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.

   The proposed change does not involve a physical change to the plant, i.e., no new or different type of equipment will be installed. The proposed change does not alter assumptions made in the safety analysis but ensures that the CST Level-low instruments perform as assumed in the [Updated Final Safety Analysis Report].

   Therefore, the proposed changes do 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 proposed change adds test requirements that will assure that (1) the CST Level-low instrumentation for the setpoint allowable value will be the limiting setting for assessing instrumentation channel operability and (2) will be conservatively determined so that the evaluation of CST instrument performance history and the requirements of the calibration procedures will not have an adverse effect on equipment operability. The testing methods and acceptance criteria for the CST Level-low instrumentation will continue to be met. There is no impact to the safety analysis acceptance criteria as described in the plant licensing basis because no change is made to the accident analysis assumptions.
>
> 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: Joseph A. Aluise, Associate General Counsel - Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: July 12, 2010.

Description of amendment request: The proposed amendment would modify Item 1 of Table 2-5, “Instrumentation Operating Requirements for Other Safety Feature Functions,” of Technical Specification (TS) 2.15, “Instrumentation and Control Systems,” to provide new Note (e), and Surveillance Requirement (SR) Items 1 and 2 of Table 3-3, “Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls,” of TS 3.1, “Instrumentation and Control,” which pertain to operability of the primary and secondary control element assembly (CEA) position indication system (CEAPIS) channels. A new SR is proposed for Item 4 of Table 3-3 of TS 3.1, which will verify the position of CEAs each shift. The proposed amendment will ensure that CEA alignment is maintained during power operations so that the power distribution and reactivity limits defined by the design power peaking and shutdown margin (SDM) limits are preserved.

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 will allow plant operation to continue when a CEAPIS channel is inoperable by requiring prompt verification of CEA positions following CEA movement. CEAs are most likely to become misaligned during movement and therefore, this change will cause CEA alignment errors to be promptly detected and corrected. It is appropriate to clarify that CEAPIS channels are not subject to the requirements of TS 2.15(1), (2), and (3) as they are not designed to be placed in trip or bypass, nor are they engineered safety feature (ESF) or isolation logic subsystems.

   The proposed amendment does not alter the requirements of TS 2.15(4) regarding the rod block function of the secondary CEAPIS channel. Should the secondary CEAPIS channel or its rod block function be inoperable, several additional CEA deviation events are possible. However, this situation is already addressed by TS 2.15(4), which requires the CEAs (rods) to be maintained fully withdrawn with the control rod drive system mode switch in the off position except when manual motion of CEA Group 4 is required to control axial power distribution. This is the same position that the CEAs must be in (fully withdrawn) when the plant is at power (Mode 1) in order to utilize distributed control system (DCS) core mimic to CHANNEL CHECK the CEAPIS channels.
   If it was not possible to use DCS core mimic to verify the primary CEAPIS channel as would be the case if CEA Group 4 was inserted to control axial power distribution, then the primary CEAPIS channel would be declared inoperable when the CHANNEL CHECK could not be accomplished. The plant would then be placed in hot shutdown (Mode 3) within 12 hours in accordance with TS 2.15(4). Therefore, although the proposed amendment

will allow a CEAPIS channel to be inoperable indefinitely, there is no significant increase in the probability or consequences of an accident as the requirements of TS 2.15(4) will continue to be met. This serves to prevent the type of CEA deviation events that the rod block function was designed for.

Replacing the current method of verifying CEAPIS data with the defined term CHANNEL CHECK is an improvement that provides additional flexibility without weakening the intent of the surveillance. As a result, when it is feasible to obtain CEA position indication from DCS core mimic (i.e., when the CEAs are either fully inserted or fully withdrawn), the primary and secondary CEAPIS channels will be compared with DCS core mimic indication as well as each other.

As an additional means of verifying CEA positions, DCS core mimic indication provides added confidence that the CEAs are in the indicated positions. Should the primary or secondary CEAPIS channel become inoperable, the accuracy and reliability of DCS core mimic indication is assured by its previous comparison with both OPERABLE channels. Comparison of the OPERABLE CEAPIS channel with DCS core mimic will satisfy the required CHANNEL CHECK and allow continued operation while the inoperable channel is repaired. The proposed amendment ensures that the CEA alignment required by TS 2.10.2(4) is met each shift by requiring all full length (shutdown and regulating) CEAs to be positioned within 12 inches of all other CEAs in the group.

The change proposed for TS 2.10.2(7)c incorporates more conservative wording to ensure that the regulating CEA groups are maintained within the Long Term Insertion Limit. The proposed change will ensure that corrective actions are taken if either time interval is exceeded and makes TS 2.10.2(7)c more consistent with CE STS.

The proposed amendment 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. As an additional means of verifying primary and secondary CEAPIS data, DCS core mimic indication increases confidence in the reliability of CEAPIS data.

The proposed amendment will help minimize unplanned shutdowns that can cause plant transients yet continues to ensure that power distribution and reactivity limits are maintained.

Therefore, it is concluded that this 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 does not change the design function or operation of the primary or secondary CEAPIS channels. If one CEAPIS channel should become inoperable, the position of CEAs will be verified within 15 minutes of any CEA movement to quickly detect and correct CEA alignment errors. Data from each CEAPIS channel will continue to be compared to the other channel each shift as before. However, a CHANNEL CHECK will require that CEAPIS channel data also be compared with DCS core mimic indication when it is available. Thus, when the CEAPIS channels are required to be OPERABLE, there will be at least two means of verifying the position of CEAs or else appropriate actions must be taken. The CEA alignment required by TS 2.10.2(4) is assured by requiring verification each shift that all full length (shutdown and regulating) CEAs are positioned within 12 inches of all other CEAs in the group.

No changes are proposed to testing and calibration of the CEAPIS channels and these requirements will continue to ensure that they are capable of performing their design function. Use of the defined term CHANNEL CHECK is an appropriate surveillance method as it requires that the channel be compared with other independent channels measuring the same variable where feasible. DCS core mimic is a diverse, accurate and reliable means of verifying CEA positions when the CEAs are fully inserted or fully withdrawn. The change proposed for TS 2.10.2(7)c ensures that appropriate corrective actions are taken when the regulating CEA groups are below the Long Term Insertion Limit in excess of either of the specified time intervals.

No new structures, systems, or components (SSCs) are being installed, and no credible new failure mechanisms, malfunctions, or accident initiators are created.

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.

   When a CEAPIS channel is inoperable, the proposed amendment allows plant operation to continue but requires more frequent verification of CEA positions following any CEA movement, which is when CEAs are most likely to become misaligned. This will enable CEA alignment errors to be detected and corrected more promptly. As CEAPIS channels are not designed to be placed in trip or bypass, nor are they engineered safety feature (ESF) or isolation logic subsystems, it is appropriate to clarify that TS 2.15(1), (2), and (3) do not apply. FCS normally operates with the CEAs fully withdrawn and maintains reactivity control by adjusting reactor coolant system (RCS) boric acid concentration. When the CEAs are fully withdrawn (or fully inserted), DCS core mimic indication provides accurate and reliable indication of CEA positions suitable for comparison with the primary and secondary CEAPIS channels. Thus, even with one CEAPIS channel inoperable, a diverse means of verifying the accuracy of the OPERABLE CEAPIS channel will be available. The accuracy and reliability of DCS core mimic is assured by testing conducted each refueling outage with continued assurance provided by comparison with primary and secondary CEAPIS each shift.

The change also ensures that the CEA alignment required by TS 2.10.2(4) is met each shift by requiring all full length (shutdown and regulating) CEAs to be positioned within 12 inches of all other CEAs in the group. The proposed amendment does not alter the TS 2.15(4) requirement to place the reactor in hot shutdown in the event that both CEAPIS channels are inoperable. The change proposed for TS 2.10.2(7)c incorporates more conservative wording to ensure that the regulating CEA groups are maintained within the Long Term Insertion Limit.

The proposed amendment will help minimize unplanned shutdowns that can cause plant transients yet continues to ensure that power distribution and reactivity limits are maintained. The proposed amendment 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, the proposed amendment 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: David A. Repka, Esq., Winston & Strawn, 1700 K Street, N.W., Washington, DC 20006-3817.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: October 4, 2010.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) requirements for snubbers in TS 3/4.7.9 due to planned revisions to the inservice inspection (ISI) program.

For the current third 10-year ISI intervals, at Salem Nuclear Generating Station (Salem), Units 1 and 2, snubber testing and examination are performed in accordance with the specific requirements of TS 3/4.7.9 in lieu of the requirements contained in American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, Article IWF-5000, as previously authorized by the U.S. Nuclear Regulatory Commission (NRC or the Commission).

Section 50.55a(g)(4)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that inservice examination of components conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b), 12 months before the start of the inspection interval. For the Salem Unit 1 fourth 10-year ISI interval beginning on May 20, 2011, the licensee intends to adopt Subsection ISTD of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2004 Edition, in place of the requirements for snubbers in ASME Code, Section XI, Articles IWF-5200(a) and (b) and IWF-5300(a) and (b), as permitted by 10 CFR 50.55a(b)(3)(v). The licensee also intends to adopt Subsection ISTD of the ASME OM Code for the remainder of the Salem Unit 2 third 10-year ISI interval which ends on November 27, 2013.

In accordance with 10 CFR 50.55a(g)(5)(ii), if a revised ISI program for a facility conflicts with the TSs for the facility, licensees are required to apply to the Commission for amendment of the TSs to conform the TSs to the revised program. Due to the planned changes to the ISI program, the proposed amendment would replace the specific TS requirements for snubbers, currently contained in surveillance requirement (SR) 4.7.9, with reference to the program for examination, testing and service life monitoring for snubbers. In addition, the current reference to SR 4.7.9c in TS ACTION 3.7.9 would be replaced with reference to the program for snubbers.

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, with Nuclear Regulatory Commission (NRC) staff edits in square brackets:

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

   Response*:* No.

   The proposed changes revise [TS 3/4.7.9 due to planned changes to the ISI program for snubbers. Specifically, the proposed amendment would replace the TS SRs for snubbers with reference to the program for examination, testing and service life monitoring for snubbers. Following implementation of the proposed amendment, in lieu of the TS SRs, snubber examination, testing and service life monitoring would be governed by the requirements in Section XI of the ASME Code or the OM Code as required by 10 CFR 50.55a(g) or 10 CFR .55a(b)(3)(v), except where the NRC has granted specific written relief, pursuant to 10 CFR 50.55a(g)(6)(i), or authorized alternatives pursuant to 10 CFR 50.55a(a)(3).]

   Snubber examination, testing and service life monitoring is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased.

   Snubbers will continue to be demonstrated OPERABLE by performance of a program for examination, testing and service life monitoring in compliance with 10 CFR 50.55a or authorized alternatives. The proposed change to TS ACTION 3.7.9 for inoperable snubbers is administrative in nature and is required for consistency with the proposed change to SR 4.7.9. Therefore the proposed change does not adversely affect plant operations, design functions or analyses that verify the capability of systems, structures, and components to perform their design functions. The consequences of accidents 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 amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response*:* No.

   The proposed changes do not involve any physical alteration of plant equipment. The proposed change does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

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

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

   Response*:* No.

   The proposed changes ensure snubber examination, testing and service life monitoring will continue to meet the requirements of 10 CFR 50.55a(g) except where the NRC has granted specific written relief, pursuant to 10 CFR 50.55a(g)(6)(i), or authorized alternatives pursuant to 10 CFR 50.55a(a)(3). Snubbers will continue to be demonstrated OPERABLE by performance of a program for examination, testing and service life monitoring in compliance with 10 CFR 50.55a or authorized alternatives. The proposed change to TS ACTION 3.7.9 for inoperable snubbers is administrative in nature and is required for consistency with the proposed change to SR 4.7.9.

   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, and with the changes noted above in square brackets, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Vincent Zabielski, PSEG Nuclear LLC - N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: September 22, 2010, as supplemented by letter dated November 22, 2010.

Description of amendment request: The proposed amendment consists of changes to the approved fire protection program as described in the Wolf Creek Generating Station (WCGS) Updated Safety Analysis Report (USAR). Specifically, amendment proposes a deviation from a commitment to certain technical requirements of 10 CFR, Part 50, Appendix R, Section III.L.1, as described in Appendix 9.5E of the WCGS USAR. The licensee has proposed to revise USAR Table 9.5E-1 to include information on Reactor Coolant System process variables not maintained within those predicted for a loss of normal ac [alternating current] power.

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

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

   Response: No.

   The design function of structures, systems and components (SSCs) are not impacted by the proposed change. Evaluation SA-08-006 Rev. 1 [RETRAN-3D Post-Fire Safe Shutdown (PFSSD) Consequence Evaluation for a Postulated Control Room Fire] has demonstrated that the formation of voids in the reactor head for a short time following a fire in the control room and spurious temporary opening of the pressurizer power operated relief valve (PORV) does not result in damage to a fission product barrier and does not result in a loss of natural circulation cooldown. The proposed change does not alter or prevent the ability of SSCs from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits.

Therefore, the probability of any accident previously evaluated is not increased. Equipment required to mitigate an accident remains capable of performing the assumed function.

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 will not alter the requirements or function for systems required during accident conditions. The design function of structures, systems and components are not impacted by the proposed change. The thermal hydraulic analysis of the reactor coolant system identified that the process variables are not maintained within those predicted for a loss of normal ac power, however, the fission product boundary integrity is not affected.

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

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

   Response: No.

   There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on departure from nuclear boiling ratio (DNBR) limits, heat flux hot channel factor ($F_Q(Z)$) limits, nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) limits, peak centerline temperature (PCT) limits, peak local power density or any other margin of safety.

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

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

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: November 4, 2010.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," to replace the existing large break loss-of-coolant accident (LOCA) analysis methodology. Specifically, the proposed change adds a reference of Westinghouse Electric Company's topical report WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," to TS 5.6.5b.

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 revises TS Section 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to TS 5.6.5b as a method used for establishing core operating limits.

Accident analyses are not accident initiators; therefore, the proposed change does not involve a significant increase in the probability of an accident. The analyses using ASTRUM demonstrated that the acceptance criteria in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for lightwater nuclear power reactors," were met. Large break LOCA analyses performed consistent with the methodology in NRC-approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met; thus, this change does not involve a significant increase in the consequences of an accident. No physical changes to the plant are associated with the proposed change.

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 change revises TS Section 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-16009-P-A to TS 5.6.5b as a method used for establishing core operating limits. There are no physical changes being made to the plant as a result of using the Westinghouse ASTRUM analysis methodology in WCAP-16009-P-A for performance of the large break LOCA analyses. Large break LOCA analyses performed consistent with the methodology in NRC-approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met. No new modes of plant operation are being introduced. The configuration, operation, and accident response of the structures or components are unchanged by use of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analyses are within the design limits of existing plant equipment.

   In addition, employing the Westinghouse ASTRUM large break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to instrumentation actuation setpoints. Adding the reference to WCAP-16009-P-A in TS Section 5.6.5b is an administrative change that does 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 accident previously evaluated.

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

   Response: No.

   The proposed change revises TS Section 5.6.5 to incorporate a new large break LOCA analysis methodology. Specifically, the proposed change adds WCAP-1 6009-P-A to TS 5.6.5b as a method used for establishing core operating limits. The analyses using ASTRUM demonstrated that the applicable acceptance criteria in 10 CFR 50.46 are met. Margins of safety for large break LOCAs include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria are not being changed by this proposed new methodology. Large break LOCA analyses performed consistent with the methodology in NRC-approved WCAP-16009-P-A, including applicable assumptions, limitations and conditions, demonstrate that 10 CFR 50.46 acceptance criteria are met; thus, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Branch Chief: Michael T. Markley.

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.

NextEra Energy Point Beach, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant (PBNP), Units 1 and 2, Manitowoc County, Wisconsin

Date of amendment request: April 7, 2009, as supplement by letters dated June 17 (two letters) and December 8 of 2009; and April 15, July 8, July 28, August 24, September 9, September 21, October 14, and November 1 of 2010.

Brief description of amendment request: The proposed amendment would increase the licensed core power level for PBNP Units 1 and 2 from 1540 to 1800 megawatts thermal. The increase in core thermal power will be approximately 17 percent over the current licensed thermal power level and is categorized as an Extended Power Uprate.

Date of publication of individual notice in *Federal Register*: November 17, 2010 (75 FR 70305).

Expiration date of individual notice: January 18, 2011.

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 01 F21, 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*.

Indiana Michigan Power Company (IandM), Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendment: September 8, 2010.

Brief description of amendment: The amendments delete the Technical Specification requirements related to the containment hydrogen recombiners and the hydrogen monitors, in accordance with Nuclear Energy Institute Technical Specification Task Force (TSTF) initiative designated as TSTF-447.

Date of issuance: December 14, 2010.

Effective date: As of the date of issuance and shall be implemented within 120 days from the date of issuance.

Amendment Nos.: 313 (for Unit 1) and 296 (for Unit 2).

Facility Operating License Nos. DPR-58 and DPR-74: Amendment revised the Renewed Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: October 14, 2010 (75 FR 63209).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 14, 2010.

No significant hazards consideration comments received: No.

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 application for amendments: November 24, 2009, as supplemented by letter dated May 26, 2010.

Brief description of amendments: These amendments revise Technical Specification (TS) 4.2.1, "Fuel Assemblies," to add Optimized ZIRLO™ as an acceptable fuel rod cladding material and add two Westinghouse topical reports to the analytical methods identified in TS 5.6.5.b.

Date of issuance: November 29, 2010.

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

Amendment Nos.: 199, 187.

Facility Operating License Nos. DPR-42 and DPR-60: Amendments revised the Technical Specifications.

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

The supplemental letter 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 amendments is contained in a Safety Evaluation dated November 29, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 16th day of December, 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

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