ATTACHMENT 1—GENERAL TARGET SCHEDULE FOR PROCESSING AND RESOLVING REQUESTS FOR ACCESS TO SENSITIVE UNCLASSIFIED NON-SAFEGUARDS INFORMATION (SUNSI) IN THIS PROCEEDING—Continued

Day	Event
205	Deadline for petitioner to seek reversal of a final adverse NRC staff determination either before the presiding officer or another designated officer.
A	If access granted: Issuance of presiding officer or other designated officer decision on motion for protective order for access to sensitive information (including schedule for providing access and submission of contentions) or decision reversing a final adverse determination by the NRC staff.
A+3	If access granted: Issuance of presiding officer or other designated officer decision on motion for protective order for access to sensitive information (including schedule for providing access and submission of contentions) or decision reversing a final adverse determination by the NRC staff.
A+28	Deadline for submission of contentions whose development depends upon access to SUNSI. However, if more than 25 days remain between the petitioner's receipt of (or access to) the information and the deadline for filing all other contentions (as established in the notice of hearing or opportunity for hearing), the petitioner may file its SUNSI contentions by that later deadline.
A+53 (Contention receipt +25)	Answers to contentions whose development depends upon access to SUNSI.
A+60 (Answer receipt +7)	Petitioner/Intervenor reply to answers.
B	Decision on contention admission.

[FR Doc. E9–11604 Filed 5–18–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2009–0204]

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 staff) 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 23, 2009, to May 6, 2009. The last biweekly notice was published on May 5, 2009 (74 FR 20741).

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 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

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

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

Written comments may be submitted by mail to the Chief, Rulemaking and Directives Branch, 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. Copies of written comments received may be examined at the Commission'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 01F21, 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 within 60 days, 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 set forth the specific contentions which the petitioner/requestor 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 petitioner/requestor 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 petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor 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 petitioner/requestor to relief. A petitioner/requestor 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, and the Commission has not made a final determination on the issue of no significant hazards consideration, 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, which the NRC promulgated in August 28, 2007 (72 FR 49139). 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 a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *hearingdocket@nrc.gov,* or by calling (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/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at *http://www.nrc.gov/site-help/e-submittals/install-viewer.html.* 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.*

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it 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 filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE 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 EIE 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 may seek assistance through the ''Contact Us'' link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html* or by calling the NRC electronic filing Help Desk, which is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays. The electronic filing Help Desk can be contacted by telephone at 1–866–672–7640 or by e-mail at *MSHD.Resource@nrc.gov.*

Participants who believe that they have a good cause for not submitting documents electronically must file a motion, 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.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

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, an Atomic Safety and Licensing Board, or a 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.

For further details with respect to this amendment action, 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 01F21, 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.* 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 e-mail to *pdr.resource@nrc.gov.*

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

Date of amendment request: February 19, 2009.

Description of amendment request: The amendments would relocate the reactor coolant system pressure and temperature (P/T) limits and the low temperature overpressure protection (LTOP) enable temperatures to a licensee-controlled document outside of the Technical Specifications (TSs). The P/T limits and LTOP enable temperatures would be specified in a Pressure and Temperature Limits Report (PTLR) that would be located in the Palo Verde Nuclear Generating Station (PVNGS) Technical Requirements Manual and administratively controlled by a new TS 5.6.9.

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 proposed change revises the Technical Specifications by relocating the reactor coolant system (RCS) pressure and temperature limits, heatup and cooldown curves and low temperature overpressure protection (LTOP) enable temperatures from the Technical Specifications to an [Arizona Public Service] APS-controlled RCS Pressure and Temperature Limits Report (PTLR), and requiring that the limits in the PTLR be determined using the analytical methods described in the NRC-approved Topical Report CE NPSD–683–A. Relocation of this information and updating it using NRC-approved methodology will not alter the requirement to update the RCS pressure and temperature curves and limits in accordance with 10 CFR 50 Appendices G and H. Updating the P/T curves and LTOP limits ensures the reactor coolant system's pressure boundary integrity is protected throughout plant life. Consequently, this proposed change is determined to not contribute to an increase in the probability of, or the initiation of, a design basis accident. Similarly, the safety analysis information presented in the Updated Final Safety Analysis Report remains unchanged.

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 revises the Technical Specifications by relocating the RCS pressure and temperature limits, heatup and cooldown curves and LTOP enable temperatures from the Technical Specifications to a PVNGS PTLR, and requiring that the limits in the PTLR be determined using the analytical methods described in the NRC-approved Topical Report CE NPSD–683–A. The PTLR documents removal, testing and analyzing the surveillance capsules, and will be updated by APS to reflect the results of testing and analysis of surveillance specimens withdrawn in the future. Removal, testing and analysis of surveillance specimens may result in a need to implement changes to the RCS pressure and temperature limits. Such changes are implemented to ensure the integrity of the RCS pressure boundary throughout plant lifetime. Updates to the RCS pressure and temperature curves and limits will not create a new or different kind of accident. Relocating the P/T curves, heatup and cooldown rates and LTOP limits to the PTLR has no impact on any safety analyses.

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.

Pressure and temperature curves and limits are provided as limits to plant operation to ensure RCS pressure boundary integrity is maintained throughout the plant's lifetime. Changes to the RCS pressure and temperature curves and limits, resulting from the removal, testing and analysis of surveillance capsules, are only made within the acceptable margin limits thereby maintaining the required margin of safety. There is no change to the safety analysis.

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 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.

Duke Energy Carolinas, LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: December 1, 2008.

Description of amendment request: The proposed amendments would correct a non-conservative Technical Specification (TS) Surveillance Requirement by revising McGuire TS 3.8.1.4 to increase the minimum required amount of fuel oil for the Emergency Diesel Generators fuel oil day tank as read on the local fuel gauge used to perform the surveillance.

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.

Implementation of the proposed amendment does not significantly increase the probability or the consequences of an accident previously evaluated. The Emergency Diesel Generators (EDGs) and their associated emergency buses function as accident mitigators. The proposed changes do not involve a change in the operational limits or the design of the electrical power systems (particularly the emergency power systems) or change the function or operation of plant equipment or affect the response of that equipment when called upon to operate. The proposed change to TS SR 3.8.1.4 confirms the minimum supply of fuel oil in the emergency diesel generators (EDG) fuel oil day tank. The minimum value for the affected parameter is being increased in the conservative direction and further ensures the EDGs ability to fulfill their safety related function. Thus, based on the above, 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 a change in the operational limits or the design capabilities of the emergency electrical power systems. The proposed changes do not change the function or operation of plant equipment or introduce any new failure mechanisms. The evaluation that supports this LAR included a review of the EDG fuel oil system to which this parameter applies. The proposed changes do not introduce any new or different types of failure mechanisms; plant equipment will continue to respond as designed and analyzed.

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

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of the fuel cladding, the reactor coolant system and the containment system will not be adversely impacted by the proposed changes. Thus, it is concluded that the proposed TS and TS Basis changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202.

NRC Branch Chief: Melanie Wong.

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

Date of amendment request: March 5, 2009.

Description of amendment request: The proposed amendment will revise the Reactor Vessel Heatup, Cooldown, and Low Temperature Overpressure Protection curves in Technical Specifications (TSs) 3.4.3 and 3.4.12 to incorporate the most recent estimates of lifetime neutron fluence and the effects of the Stretch Power Uprate (Amendment No. 241).

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

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

The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the heatup and cooldown limitation curves. The proposed changes do not modify the RCS [Reactor Coolant System] pressure boundary. That is, there are no changes in operating pressure, materials, or seismic loading. The proposed changes do not adversely affect the integrity of the RCS pressure boundary such that its function in the control of radiological consequences is affected. The proposed heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10 CFR 50 [Title 10 of the Code of Federal Regulations Part 50] Appendix G, and ASME B&PV code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code], Section XI, Appendix G edition with 2000 Addenda. The proposed heatup and cooldown limitation curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of RPV [Reactor Pressure Vessel] beltline materials. Use of this methodology provides compliance with the intent of 10 CFR 50 Appendix G and provides margins of safety that ensure non-ductile failure of the RPV will not occur. The proposed heatup and cooldown limitation curves prohibit operation in regions where it is possible for non-ductile failure of carbon and low alloy RCS materials to occur. Hence, the primary coolant pressure boundary integrity will be maintained throughout the limit of applicability of the curves, 29.2 EFPY [Effective Full-Power Years].

Operation within the proposed LTOPS [Low Temperature Overpressure Protection System] limits ensures that overpressurization of the RCS at low temperatures will not result in component stresses in excess of those allowed by the ASME B&PV Code Section XI Appendix G.

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

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

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

Consequently, the proposed changes do not involve a significant increase in the probability or consequence of a new or different kind of accident, from any accident previously evaluated.

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

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

The revised heatup and cooldown limitation curves and LTOPS limits are established in accordance with current regulations and the ASME B&PV Code 1998 edition with 2000 Addenda. These proposed changes are acceptable because the ASME B&PV Code maintains the margin of safety required by 10 CFR 50.55(a). Because operation will be within these limits, the RCS materials will continue to behave in a non-brittle manner consistent with the original design bases.

The proposed changes to the allowable operation of charging and safety injection pumps when LTOPS is required to be operable is consistent with the IP2 licensing bases as established in TS Amendment 224.

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

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

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

NRC Acting Branch Chief: Richard V. Guzman.

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

Date of amendment request: March 25, 2009.

Description of amendment request: The proposed amendment would add two Emergency Core Cooling System (ECCS) valves to Surveillance Requirement (SR) 3.5.2.1. The SR is designed to verify that ECCS valves whose single failure could cause loss of the ECCS function are in the required position with ac power removed so that misalignment or single failure cannot prevent completion of the ECCS function. Entergy plans to install an alternate source of power during the spring 2010 refueling outage to provide the required position indication. The proposed changes support Entergy's resolution to Generic Letter (GL) 2004–02 by establishing a licensing basis that supports meeting the regulatory requirements of the GL.

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 two ECCS valves to SR 3.5.2.1. The purpose of the surveillance is to assure that the valves are in their required position with normal ac power removed from the valve operator so that misalignment or single failure cannot prevent completion of the ECCS function. The performance of the SR does not involve any actions related to the initiation of an accident and therefore the proposed changes cannot increase the probability of an accident. Misalignment or single failure of one of the two valves being added to TS [Technical Specifications] could cause a loss of the ECCS function based on GSI [Generic Safety Issue]-191 evaluations, so the change will not increase the consequences of an accident but rather provide assurance that no such increase can occur. 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 adds two ECCS valves to SR 3.5.2.1. The purpose of the surveillance is to assure that the valves are in their required position with normal ac power removed from the valve operators so that misalignment or single failure cannot prevent completion of the ECCS function. The provision of alternate power to the existing valve position indication during the upcoming spring 2010 outage (2R19), will allow the valve operators to be normally deenergized. The change assures that the valves will be in their correct position and does not introduce any new failure modes or the possibility of a different 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 change involve a significant reduction in a margin of safety?

Response—No.

The proposed change adds two ECCS valves to SR 3.5.2.1. The purpose of the surveillance is to assure that the valves are in their required position with normal ac power removed so that misalignment or single failure cannot prevent completion of the ECCS function. The valves will be re-energized 24 hours following a DBA [design-basis accident] and therefore will be capable of performing their required function of isolating a potential passive failure at that time. This ensures that the ECCS function can be performed without a reduction in the margin of safety.

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

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

NRC Acting Branch Chief: Richard V. Guzman.

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

Date of amendment request: March 29, 2009.

Description of amendment request: The proposed amendment will establish a more restrictive acceptance criterion for surveillance requirement (SR) 3.8.6.6 regarding periodic verification of capacity for the affected station batteries.

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 acceptance criterion applied to an existing surveillance test for the Indian Point 2 station batteries. Performing a technical specification surveillance test is not an accident initiator and does not increase the probability of an accident occurring. The proposed revision to the test acceptance criterion is based on the design calculation for battery performance at the minimum design temperature. The proposed new value for the test acceptance criteria is more limiting than the existing value which does not account for the minimum environmental design temperature assumed for the limiting battery locations. Establishing a test acceptance criterion that bounds existing or assumed conditions validates the equipment performance assumptions used in the accident mitigation safety analyses. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed change revises the test acceptance criterion for an existing technical specification surveillance test conducted on the existing station batteries. 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 in test acceptance criterion 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 conduct of performance tests on safety-related plant equipment is a means of assuring that the equipment is capable of performing its intended safety function and therefore maintaining the margin of safety established in the safety analysis for the facility. The proposed change in the acceptance criterion for the battery capacity surveillance test is more conservative and more restrictive than the value currently in the technical specification and is based on the applicable design calculation for these components.

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

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

NRC Acting Branch Chief: Richard V. Guzman.

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

Date of amendment request: March 26, 2009.

Description of amendment request: The proposed amendments would revise the fire protection program (FPP) to eliminate the requirement for the backup manual carbon dioxide (CO_2) fire suppression system in the upper cable spreading rooms.

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 revises the FPP to eliminate the requirement to maintain the backup CO_2 fire suppression system for the upper cable spreading rooms. With the exception of the CO_2 fire suppression system itself, the proposed change does not result in any physical changes to safety related structures, systems, or components [SSCs], or the manner in which they are operated, maintained, modified, tested, or inspected. The proposed change does not degrade the performance or increase the challenges of any safety related SSCs assumed to function in the accident analysis. The proposed change does not change the probability of a fire occurring since the fire ignition frequency is independent of the method of fire suppression. The proposed change does not affect the consequences of an accident previously evaluated since the fire safe shutdown analysis assumes fire damage throughout the affected fire area. The results of a fire in the upper cable spreading room would only affect one engineered safety features division. Sufficient redundancy exists in the engineered safety features fed from the other division to achieve a reactor shutdown and to maintain the reactor in a safe shutdown condition.

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 revises the FPP to eliminate the requirement to maintain the backup CO_2 fire suppression system for the upper cable spreading rooms. With the exception of the CO_2 fire suppression system itself, the proposed change does not result in any physical changes to safety related structures, systems, or components, or the manner in which they are operated, maintained, modified, tested, or inspected. The proposed change does not degrade the performance or increase the challenges of any safety related SSCs assumed to function in the accident analysis. As a result, the proposed change does not introduce nor increase the number of failure mechanisms of a new or different type than those previously evaluated. The fire safe shutdown analysis assumes fire damage throughout the area consistent with a complete lack of fire suppression capability. Potential habitability hazards associated with actuation of the CO_2 system are eliminated with the proposed change.

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

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

Response: No.

The proposed change revises the FPP to eliminate the requirement to maintain the backup CO_2 fire suppression system for the upper cable spreading rooms. With the exception of the CO_2 fire suppression system itself, the proposed change does not result in any physical changes to safety related structures, systems, or components, or the manner in which they are operated, maintained, modified, tested, or inspected. The proposed change does not degrade the performance or increase the challenges of any safety related SSCs assumed to function in the accident analysis. Since the backup manual CO_2 fire suppression system is not credited in the safe shutdown analysis to protect the upper cable spreading rooms, the proposed change does not impact plant safety since the conclusions of the fire safe shutdown analysis remain unchanged.

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

The Nuclear Regulatory Commission (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: Russell Gibbs.

Luminant Generation Company LLC, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: April 1, 2009.

Brief description of amendments: The proposed amendment would delete Technical Specification (TS) 5.2.2.d, in TS 5.2.2, ''Unit Staff,'' regarding the requirement to develop and implement administrative procedures to limit the working hours of personnel who perform safety-related functions. The requirements of TS 5.2.2.d have been superseded by Title 10 of the Code of Federal Regulations (10 CFR) Part 26, Subpart I. The change is consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Revision 0 to Technical Specification Task Force (TSTF) Improved Technical Specification Change Traveler, TSTF–511, ''Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26.''

The NRC staff issued a ''Notice of Availability of Model Safety Evaluation, Model No Significant Hazards Determination, and Model Application for Licensees That Wish to Adopt TSTF–511, Revision 0, 'Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26,' '' in the **Federal Register** on December 30, 2008 (73 FR 79923). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated April 1, 2009, 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 adopted by the licensee, is presented below:

Criterion 1: The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change removes TS restrictions on working hours for personnel who perform safety related functions. The TS restrictions are superseded by the worker fatigue requirements in 10 CFR 26. Removal of the TS requirements will be performed concurrently with the implementation of the 10 CFR 26, Subpart I, requirements. The proposed change does not impact the physical configuration or function of plant structures, systems, or components (SSCs) or the manner in which SSCs are operated, maintained, modified, tested, or inspected. Worker fatigue is not an initiator of any accident previously evaluated. Worker fatigue is not an assumption in the consequence mitigation of any accident previously evaluated.

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

Criterion 2: The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from Any Accident Previously Evaluated

The proposed change removes TS restrictions on working hours for personnel who perform safety related functions. The TS restrictions are superseded by the worker fatigue requirements in 10 CFR 26. Working hours will continue to be controlled in accordance with NRC requirements. The new rule allows for deviations from controls to mitigate or prevent a condition adverse to safety or as necessary to maintain the security of the facility. This ensures that the new rule will not unnecessarily restrict working hours and thereby create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not alter the plant configuration, require new plant equipment to be installed, alter accident analysis assumptions, add any initiators, or effect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected.

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

Criterion 3: The Proposed Change Does Not Involve a Significant Reduction in a Margin of Safety

The proposed change removes TS restrictions on working hours for personnel who perform safety related functions. The TS restrictions are superseded by the worker fatigue requirements in 10 CFR 26. The proposed change does not involve any physical changes to plant or alter the manner in which plant systems are operated, maintained, modified, tested, or inspected. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not 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.

Removal of plant-specific TS administrative requirements will not reduce a margin of safety because the requirements in 10 CFR 26 are adequate to ensure that worker fatigue is managed.

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

NRC Branch Chief: Michael T. Markley.

Luminant Generation Company LLC, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: April 2, 2009.

Brief description of amendments: The amendment revises Technical Specification (TS) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' to add Surveillance Requirement (SR) 3.3.1.16 to Function 3 of TS Table 3.3.1–1. SR 3.3.1.16 requires that RTS RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 3 is the power range neutron flux—high positive rate reactor trip function (hereafter referred to as the positive flux rate trip (PFRT) function). This change is based on a reanalysis of the Rod Cluster Control Assembly Bank Withdrawal at Power event.

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

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

Response: No.

The proposed change imposes additional surveillance requirements to assure safety related structures, systems, and components are 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 positive flux rate trip (PFRT) function.

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 flux 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 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 changes will not alter any assumptions or change any mitigation actions in the radiological consequences evaluations in the updated Final Safety Analysis Report (FSAR).

The proposed changes do 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 or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

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 accident previously evaluated?

Response: No.

The proposed change imposes additional surveillance requirements to assure safety related structures, systems, and components are verified to be consistent with the safety analysis and licensing basis.

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 or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance requirements. The additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

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

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

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

Response: No.

The proposed change imposes additional surveillance requirements to assure safety related structures, systems, and components are verified to be consistent with the safety analysis and licensing basis.

The proposed changes do not affect the acceptance criteria for any analyzed event. The margin of safety is affected in that in the new analyses of the Rod (Bank) Withdrawal at Power analyses, it is necessary to credit a previously uncredited reactor trip function in an analysis. However, that reactor trip function is described in the plant Technical

Specifications with well-defined operability requirements. An additional attribute, specifically the channel response time verification on, a periodic frequency, provides additional assurance that the trip function performs as credited in the accident analysis. With the credit for this reactor trip function, all relevant event acceptance criteria continue to be met. None of the event acceptance limits are exceeded, and none of the event acceptance limits are revised by the proposed activity. There is no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor is there any effect on those plant systems necessary to assure the accomplishment of protection functions. There is no impact on the overpower limit, the minimum departure from nucleate boiling ratio limit, the radial and axial peaking factor limits, the loss of coolant accident (LOCA) peak clad temperature limit, nor any other limit which, in whole or in part, defines a margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

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

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

NRC Branch Chief: Michael T. Markley.

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1 (NMP 1), Oswego County, New York

Date of amendment request: March 3, 2009.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Section 3.2.1, ''Reactor Vessel Heatup and Cooldown Rates,'' and Section 3.2.2, ''Minimum Reactor Vessel Temperature for Pressurization,'' by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature limit curves with references to the Pressure and Temperature Limits Report (PTLR). In addition, a new definition for the PTLR would be added to TS Section 1.0, ''Definitions,'' and a new section addressing administrative requirements for the PTLR would be added to TS Section 6.0, ''Administrative Controls.'' The proposed changes are consistent with the guidance in Generic Letter 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,'' as supplemented by TS Task Force (TSTF) traveler TSTF–419–A, ''Revise PTLR Definition and References in ISTS 5.6.6, RCS [Reactor Coolant System] PTLR.''

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 changes modify the TS by replacing references to existing reactor vessel heatup and cooldown rate limits and P–T [pressure-temperature] limit curves with references to the PTLR. The proposed amendment also adopts the NRC-approved methodology of SIR–05–044–A for the preparation of NMP1 P–T limit curves. In 10 CFR 50 Appendix G, requirements are established to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Implementing the NRC-approved methodology for calculating P–T limit curves and relocating those curves to the PTLR provide an equivalent level of assurance that RCPB integrity will be maintained, as specified in 10 CFR 50 Appendix G.

The proposed changes do not adversely affect accident initiators or precursors, and do not alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety function is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

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

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

Response: No.

The change in methodology for calculating P–T limits and the relocation of those limits to the PTLR do not alter or involve any design basis accident initiators. RCPB integrity will continue to be maintained in accordance with 10 CFR 50 Appendix G, and the assumed accident performance of plant structures, systems and components will not be affected. These changes do not involve any physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed), and installed equipment is not being operated in a new or different manner. Thus, no new failure modes are introduced.

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 amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not affect the function of the RCPB or its response during plant transients. By calculating the P–T limits using NRC-approved methodology, adequate margins of safety relating to RCPB integrity are maintained. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined, there are no changes to the setpoints at which actions are initiated, and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Acting Branch Chief: John P. Boska.

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2 (NMP 2), Oswego County, New York

Date of amendment request: March 9, 2009.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) testing frequency for the surveillance requirement (SR) in TS 3.1.4, ''Control Rod Scram Times.'' Specifically, the proposed change is based on TS Task Force (TSTF) change traveler TSTF–460–A, Revision 0, and extends the frequency for testing control rod scram time testing in SR 3.1.4.2 from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. A notice of availability of this proposed TS change using the consolidated line item improvement process was published in the **Federal Register** on August 23, 2004 (69 FR 51864). The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

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 extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The frequency of surveillance testing is not an initiator of any accident previously evaluated. The frequency of surveillance testing does not affect the ability to mitigate any accident previously evaluated, as the tested component is still required to be operable.

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 extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change does not result in any new or different modes of plant operation.

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

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change continues to test the control rod scram time to ensure the assumptions in the safety analysis are protected.

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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Acting Branch Chief: John P. Boska.

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

Date of amendment request: March 5, 2009, as supplemented by letter dated April 13, 2009.

Description of amendment request: The proposed amendments would make changes to the PINGP Technical Specifications (TSs) to revise TS 3.8.1, ''AC Sources—Operating,'' Surveillance Requirement (SR) 3.8.1.8 Frequency to allow use of the SR 3.0.2 interval extension (1.25 times the specified 24 month Frequency). This would be an exception to the SR 3.0.2 limitations in the PINGP TS, which do not allow use of the interval extension for SRs with a 24 month Frequency.

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 add a Frequency Note to Surveillance Requirement 3.8.1.8 which will allow application of the Surveillance Requirement 3.0.2 interval extension (1.25 times the specified 24 month Frequency) for performance of this surveillance. This would be an exception to the limitations specified in the Prairie Island Nuclear Generating Plant Technical Specification Surveillance Requirement 3.0.2 for Surveillance Requirements with a 24 month Frequency and would allow an interval up to 30 months for performance of the surveillance.

The emergency diesel generators are not accident initiators and therefore, these changes do not involve a significant increase [in] the probability of an accident.

Failure of the bypass relay, by itself, does not prevent an emergency diesel generator from performing its safety related functions. Since the accident analyses only require one of the two trains of onsite emergency AC to be operable, the changes proposed in the license amendment request do not involve a significant increase in the consequences of an accident.

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

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

Response: No.

This license amendment request proposes to add a Frequency Note to Surveillance Requirement 3.8.1.8 which will allow application of the Surveillance Requirement 3.0.2 interval extension (1.25 times the specified 24 month Frequency) for performance of this surveillance. This would be an exception to the limitations specified in the Prairie Island Nuclear Generating Plant Technical Specification Surveillance Requirement 3.0.2 for Surveillance Requirements with a 24 month Frequency and would allow an interval up to 30 months for performance of the surveillance.

The changes proposed for the emergency diesel generators do not change any system operations or maintenance activities. Testing requirements will be revised and will continue to demonstrate that the Limiting Conditions for Operation are met and the system components are functional. The revised test Frequency does not create new failure modes or mechanisms and no new accident precursors are generated.

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 amendment involve a significant reduction in a margin of safety?

Response: No.

This license amendment request proposes to add a Frequency Note to Surveillance Requirement 3.8.1.8 which will allow application of the Surveillance Requirement 3.0.2 interval extension (1.25 times the specified 24 month Frequency) for performance of this surveillance. This would be an exception to the limitations specified in the Prairie Island Nuclear Generating Plant Technical Specification Surveillance Requirement 3.0.2 for Surveillance Requirements with a 24 month Frequency and would allow an interval up to 30 months for performance of the surveillance.

The proposed change will continue to ensure that the DG trips bypass function operates as designed. The functionality and operability of the emergency power system is not being changed. Since the requested change only allows extension of the relay testing interval and failure of the relay by itself does not prevent the diesel from performing its safety function, this change does not involve a significant reduction in a margin of safety.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment 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: Lois M. James.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: March 30, 2009.

Description of amendment request: The proposed amendment revises the Technical Specifications (TS), Appendix A to Facility Operating License Nos. NPF–2 and NPF–8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. The changes would eliminate the Reactor Coolant Pump (RCP) Breaker Position reactor trip. The changes will allow the elimination of a trip circuitry that is susceptible to single failure vulnerabilities which can result in unwarranted reactor trips.

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

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

Response: No.

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the Final Safety Analysis Report (FSAR). All of the safety analyses have been evaluated for impact. The elimination of Reactor Coolant Pump Breaker Position reactor trip will not initiate any accident; therefore, the probability of an accident has not been increased. An evaluation of dose consequences, with respect to the proposed changes, indicates there is no impact due to the proposed changes and all acceptance criteria continue to be met. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The changes have no adverse effects on any safety-related system. Therefore, all accident analyses criteria continue to be met and these 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 a margin of safety?

Response: No.

The proposed changes do not involve a significant reduction in a margin of safety. All analyses that credit the Reactor Coolant System Low Flow reactor trip function have been reviewed and no changes to any inputs are required. The evaluation demonstrated that all applicable acceptance criteria are met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the preceding evaluation, SNC has determined that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Branch Chief: Melanie C. Wong.

Tennessee Valley Authority (TVA), Docket No. 50 390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: April 30, 2009.

Description of amendment request: The proposed amendment would revise technical specification (TS) Section 5.7, ''Procedures, Programs, and Manuals,'' to correct typographical errors introduced in Amendment No. 70, dated October 8, 2008.

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. This change is limited only to correcting a typographical error in a section number (5.7.2.20 versus 5.2.7.20) contained in Technical Specification Section 5.0, which will not change the intent of the added section previously approved in License Amendment 70. Therefore, no increase in the probability or consequences of an accident previously evaluated has been created.

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

Response: No. This change is limited only to correcting a typographical error in a section number (5.7.2.20 versus 5.2.7.20) contained in Technical Specification Section 5.0, which will not change the intent of the added section previously approved in License Amendment 70. Therefore, the possibility of a new or different kind of accident from those previously analyzed has not been created.

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

Response: No. This change is limited only to correcting a typographical error in a section number (5.7.2.20 versus 5.2.7.20) contained in Technical Specification Section 5.0, which will not change the intent of the added section previously approved in License Amendment 70. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA 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 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: L. Raghavan.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Unit Nos. 1 and 2, Louisa County, Virginia

Date of amendment request: March 26, 2009.

Description of amendment request: The proposed amendments would increase each unit's rated thermal power (RTP) level from 2893 megawatts thermal (MWt) to 2940 MWt, and make technical specification changes as necessary to support operation at the uprated power level. The proposed change is an increase in RTP of approximately 1.6 percent.

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 will increase the North Anna Power Station (NAPS) Units 1 and 2 rated thermal power (RTP) from 2893 megawatts thermal (MWt) to 2940 MWt. Nuclear steam supply systems and balance-of-plant systems, components and analyses that could be affected by the proposed change to the RTP were evaluated using revised design parameters. The evaluations determined that these structures, systems and components are capable of performing their design function at the proposed uprated RTP of 2940 MWt. An evaluation of the accident analyses demonstrates that the applicable analysis acceptance criteria are still met with the proposed changes. Power level is an input assumption to equipment design and accident analyses, but it is not a transient or accident initiator. Accident initiators are not affected by the power uprate, and plant safety barrier challenges are not created by the proposed changes.

The radiological consequences of operation at the uprated power conditions have been assessed. The proposed change to RTP does not affect release paths, frequency of release, or the analyzed source term for any accidents previously evaluated in the NAPS Updated Final Safety Analysis Report. Structures, systems and components required to mitigate transients are capable of performing their design functions with the proposed changes, and are thus acceptable. Analyses performed to assess the effects of mass and energy releases remain valid. The source term used to assess radiological consequences was reviewed and determined to bound operation at the proposed power level.

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

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

Response: No.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any proposed changes. The UFM has been analyzed, and system failures will not adversely affect any safety-related system or any structures, systems or components required for transient mitigation. Structures, systems and components previously required for transient mitigation are still capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at RTP of 2940 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

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

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

Response: No.

The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant system pressure boundary, and containment barriers. Core analyses demonstrate that power uprate implementation will continue to meet the current nuclear design basis. Impacts to components associated with the reactor coolant system pressure boundary structural integrity, and factors such as pressure-temperature limits, vessel fluence, and pressurized thermal shock were determined to be bounded by the current analyses.

Systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following implementation of the proposed change. The current NAPS safety analyses, including the design basis radiological accident dose calculations, bound the power uprate.

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

NRC Branch Chief: Melanie C. Wong.

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 Systems (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 e-mail to *pdr.resource@nrc.gov.*

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station (KPS), Kewaunee County, Wisconsin

Date of application for amendment: September 11, 2008, as supplemented by letter dated December 17, 2008, and January 20, 2009.

Brief Description of amendment: The amendment revised the Technical Specifications, extending the 15-year interval between containment Type A tests specified by Specification 4.4.a, ''Integrated Leak Rate Test,'' by 6 months. The current Type A test interval expires at the end of April 2009. The amendment extends this interval, on a one-time basis, to October 2009 to coincide with completion of the next scheduled refueling outage.

Date of issuance: April 27, 2009.

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

Amendment No.: 204.

Facility Operating License No. DPR–43: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: November 4, 2008 (73 FR 65689). The commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 2009.

No significant hazards consideration comments received: No.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station (KPS), Kewaunee County, Wisconsin

Date of application for amendment: July 7, 2008, as supplemented on September 19, 2008, and March 17, 2009.

Brief description of amendment: The amendment revised the licensing basis, authorizing the licensee to use the methodology conveyed in the licensee's letters cited above to determine the seismic loads on the recently upgraded Auxiliary Building crane. The authorization is conveyed by addition of a new License Condition 2.C.(11) to Facility Operating License DPR–43.

Date of issuance: April 30, 2009.

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

Amendment No.: 205.

Facility Operating License No. DPR–43: The amendment revised Facility Operating License No. DPR–43.

Date of initial notice in ***Federal Register***: August 26, 2008 (73 FR 50358). The Commission's related evaluation of the amendment is contained in a safety evaluation dated April 30, 2009.

No Significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: July 16, 2008, as supplemented by letters dated January 2 and March 19, 2009.

Brief description of amendment: The amendment revised Technical Specifications 3.1.4, ''Control Rod Scram Times,'' 3.2.2, ''Minimum Critical Power Ratio (MCPR),'' and 5.6.3, ''Core Operating Limits Report (COLR),'' to allow incorporation of the analytical methodologies associated with operation of Global Nuclear Fuel-Americas (GNF) fuel into the licensing basis to support transition to GNF GE14 fuel.

Date of issuance: May 5, 2009.

Effective date: As of its date of issuance and shall be implemented prior to beginning operating cycle 20.

Amendment No.: 211.

Facility Operating License No. NPF–21: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register***: October 14, 2008 (73 FR 60729).

The supplements dated January 2 and March 19, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 18, 2008, as supplemented by letter dated February 4, 2009.

Brief description of amendment: The amendment modified Technical Specification (TS) requirements for inoperable snubbers by relocating the current TS 3.7.8, ''Snubbers,'' to the Technical Requirements Manual and adding Limiting Condition for Operation (LCO) 3.0.8. The amendment also made conforming changes to TS LCO 3.0.1. The proposed amendment is consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF–372, Revision 4, ''Addition of LCO 3.0.8, Inoperability of Snubbers,'' as part of the consolidated line item improvement process.

Date of issuance: May 1, 2009.

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

Amendment No.: 219.

Facility Operating License No. NPF–38: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** December 16, 2008 (73 FR 76410). The supplemental letter dated February 4, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 1, 2009.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of application for amendment: November 18, 2008.

Brief description of amendment: This amendment modifies Technical Specification 5.5.6 to incorporate Technical Specification Task Force (TSTF) Travelers TSTF–479, ''Changes to Reflect Revision of 10 CFR [Code of Federal Regulations] 50.55a,'' and TSTF–497, ''Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less.''

Date of issuance: May 1, 2009.

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

Amendment No.: 151.

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

Date of initial notice in ***Federal Register:*** January 27, 2009 (74 FR 4772). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 1, 2009.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: June 11, 2008, as supplemented by letters dated September 15, 2008, December 10, 2008, and March 16, 2009.

Brief description of amendment: The amendment deletes Technical Specification 6.5, which provided the requirements related to review and audit functions.

Date of issuance: May 1, 2009.

Effective date: May 1, 2009.

Amendment No.: 63.

Possession Only License No. DPR–73: The amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 26, 2008 (73 FR 50356) The Commission's related evaluation of the amendment is contained in a Safety Evaluation Report, dated May 1, 2009.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 27, 2007, as supplemented on April 28, September 4, and December 17, 2008.

Brief description of amendment: The amendment revises surveillance requirements in Technical Specifications (TS) Section 3.8.1, ''AC Sources—Operating,'' associated with the diesel generator (DG) steady-state frequency and voltage. The amendment corrects non-conservative TS frequency and voltage values, which the licensee states have the potential to result in undesirable effects such as centrifugal charging pump motor brake horsepower exceeding its nameplate maximum horsepower, and subsequently overloading the DGs.

Date of issuance: April 30, 2009.

Effective date: As of the date of issuance and shall be implemented within 45 days from the date of issuance April 30, 2009.

Amendment Nos.: 309 (CNP–1), 291 (CNP–2).

Facility Operating License Nos. DPR–58 and DPR–74: Amendment revises the Renewed Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** November 4, 2008 (73 FR

65696). The April 28 and December 17, 2008 supplements provided additional information that clarified the application, but did not expand the scope of the application as originally noticed, and did not change the staff's original proposed significant hazards consideration published in the **Federal Register** on August 14, 2007.

The September 4, 2008 supplement provided additional information which expanded the scope of the application as originally noticed. The NRC staff identified that the specified DG voltage of 3,740 volts at 10 seconds after the DG start was non-conservative and inconsistent with the 3,910 volt minimum steady-state voltage provided in other parts of TS Section 3.8.1. The licensee proposed additional changes to TS Section 3.8.1 in its September 4, 2008 letter. The NRC staff determined that the proposed expanded scope of the amendment involved a proposed no significant hazards consideration as published in the **Federal Register** on November 4, 2008 (73 FR 65696).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 30, 2009.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: February 24, 2009.

Brief description of amendments: The amendments deleted the requirement for the power range neutron flux rate-high negative rate trip (Function 3.b) in Technical Specification (TS) Table 3.3.1–1, ''Reactor Trip System Instrumentation.'' The changes are consistent with the NRC-approved methodology presented in Westinghouse Topical Report, WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event,'' dated January 1990. The amendments also incorporated editorial changes to reflect the deletion of Function 3.b in TS Table 3.3.1–1.

Date of issuance: April 29, 2009.

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

Amendment Nos.: Unit 1—205; Unit 2—206.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Facility Operating Licenses and Technical Specifications.

Date of initial notice in ***Federal Register:*** March 24, 2009 (74 FR 12394).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 2009.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant (WBN), Unit 1, Rhea County, Tennessee

Date of application for amendment: December 31, 2008 superseded the application dated August 1, 2008, as supplemented by letters dated November 25 and December 31, 2008.

Brief description of amendment: The amendment revised WBN Unit 1 Technical Specification (TS) 4.2.1, ''Fuel Assemblies,'' and TS surveillance requirements (SRs) 3.5.1.4, ''Accumulators,'' and 3.5.4.3, ''RWST [Refueling Water Storage Tank],'' to increase the maximum number of tritium producing burnable absorber rods from 400 to 704.

Date of issuance: April 30, 2009.

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

Amendment No.: 77.

Facility Operating License No. NPF–90: Amendment revises the TS 4.2.1 and TS SRs 3.5.1.4 and 3.5.4.3.

Date of initial notice in ***Federal Register:*** Originally November 12, 2008 (73 FR 66946) was superseded by a notice on January 27, 2009 (74 FR 4776).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 30, 2009.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 7th day of May 2009.

For the Nuclear Regulatory Commission.

Joseph G. Giitter,

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

[FR Doc. E9–11268 Filed 5–18–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 70–1113; NRC–2009–0209]

Notice of Availability of Environmental Assessment and Finding of No Significant Impact for License Renewal for Global Nuclear Fuel—Americas, LLC, Wilmington, NC

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

FOR FURTHER INFORMATION CONTACT: Mary Adams, Senior Project Manager, Fuel Manufacturing Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Rockville, Maryland 20852. *Telephone:* (301) 492–3113; *Fax:* (301) 492–3363; *e-mail: Mary.Adams@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

The Nuclear Regulatory Commission (NRC) is considering the renewal of Special Nuclear Material License SNM–1097 for the continued operation of the Global Nuclear Fuel—Americas, LLC (GNF–A) Fuel Fabrication Facility located in Wilmington, North Carolina. This renewal authorizes the licensee to receive and possess nuclear materials at the Wilmington facility to fabricate and assemble nuclear fuel components under the provisions of 10 CFR Part 70, Domestic Licensing of Special Nuclear Material. If NRC approves the renewal of the license, the term would cover 40 years. NRC has prepared an environmental assessment (EA) in support of this action in accordance with the requirements of 10 CFR Part 51. Based on the EA, the NRC has concluded that a finding of no significant impact is appropriate. If approved, NRC will issue the renewed license following the publication of this Notice.

II. EA Summary

The licensee requests approval to renew SNM–1097 for an additional 40 years at the Wilmington, North Carolina facility. Specifically, this would allow GNF–A to continue manufacturing and assembling nuclear fuel components for use in commercial light-water-cooled nuclear power reactors. GNF–A's request for the renewal was previously noticed in the **Federal Register** on June 18, 2007 (72 FR 33539), with an opportunity to request a hearing. No hearing requests were received.

The staff has prepared the EA in support of the proposed license renewal. Staff considered direct, indirect, and cumulative environmental impacts to 12 resource areas in their evaluation, including: land use; transportation; socioeconomics; air quality; water quality; geology and soils; ecology; noise; historic and cultural; scenic and visual; public and occupational health; and waste management. All of the environmental impacts were small-to-moderate. The license renewal request does not require altering the site footprint nor does it