7590-01-P

NUCLEAR REGULATORY COMMISSION

BIWEEKLY NOTICE

[NRC-2010-0017]

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 31, 2009, to January 13, 2010. The last biweekly notice was published on January 12, 2010 (75 FR 1655).

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

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

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

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

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

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

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/doc-collections/cfr/. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

To comply with the procedural requirements of E-Filing, at least ten (10) days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and

(2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

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

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

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than

11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

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

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants.

Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

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

Petitions for leave to intervene must be filed no later than 60 days from **[INSERT DATE OF PUBLICATION IN FEDERAL REGISTER].** Non-timely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

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: October 30, 2009.

Description of amendment request: The amendments would revise License Condition C.(1) for Units 1 and 3, and the Technical Specifications (TS) for all three units, to remove requirements no longer applicable due to the completion of power uprate, replacement of steam generators, removal of part-length control element assemblies (CEAs), and completion of a core protection calculator (CPC) upgrade, and to make a minor administrative change to the nomenclature of the containment sump trash racks and screens.

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

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

   Response: No.

   The proposed amendment includes the following changes that are considered to be administrative and/or editorial changes:

   A. Remove superseded references to 3876 megawatts thermal (MWt) and related information to this value from Unit 1 and Unit 3 Operating Licenses and Unit 1, 2, and 3 Technical Specifications.

      This change is administrative. The change only removes the references to 3876 MWt and related information to this value and leaves the references to 3990 MWt.

   B. Remove references to Part Length Control Element Assemblies.

This change is administrative because it only removes references to part length CEAs which have been replaced by part strength CEAs.

C. Remove outdated pages and other references as a result of the CPC upgrade, and adjust the indentation of the logical connectors AND and OR in TS 3.2.4, between Required Actions B.1, B.2.1, and B.2.2.

This change is administrative because it removes the redundant TS pages identified as "(Before CPC Upgrade) or (Before CPCS Upgrade)" and removes the reference to "(After CPC Upgrade) or (After CPCS Upgrade)" from various TS pages that will be renumbered and remain in place. The CPC upgrade has been completed. The adjustment of the indentation of the logical connectors AND and OR in TS 3.2.4 is consistent with the Action numbers and with TS 1.2.

D. Change "trash racks and screens" to "strainers."

This change is administrative. The change from "trash racks and screens" to "strainers" does not change the intent of the Surveillance Requirement 3.5.3.8 to verify, by visual inspection, that each [emergency core cooling system] ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.

E. Delete inspection requirements for Steam Generators (SG) with Alloy 600 MA tubes.

This change is administrative because APS [Arizona Public Service Company] has completed the SG replacement project which removed all SGs containing Alloy 600 MA tubes.

As discussed above, the proposed amendment involves administrative and/or editorial changes only. The proposed amendment does not impact any accident initiators, analyzed events, or assumed mitigation of accident or transient events. The proposed changes do not involve the addition or removal of any equipment or any design changes to the facility. The proposed changes do not affect any plant operations, design function, or analysis that verifies the capability of structures, systems, and components (SSCs) to perform a design function. The proposed changes do not change any of the accidents previously evaluated in the UFSAR [updated final safety analysis report]. The proposed changes do not affect SSCs, operating procedures, and administrative controls that have the function of preventing or mitigating any of these accidents.

Therefore, the proposed changes do not represent 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.

   As stated in response to standard 1, the proposed amendment only involves administrative and/or editorial changes. No actual plant equipment or accident analyses will be affected by the proposed changes. The proposed changes will not change the design function or operation of any SSCs. The proposed changes will not result in any new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. The proposed amendment does not impact any accident initiators, analyzed events, or assumed mitigation of accident or transient events. Therefore, this proposed change does not create the possibility of an accident of a new or different kind than previously evaluated.

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

   Response: No.

   As stated in response to standard 1, the proposed amendment only involves administrative and/or editorial changes. The proposed change does not involve any physical changes to the 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. 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.

Carolina Power & Light Company, Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: October 27, 2009.

Description of amendments request: The proposed amendments would modify technical specifications (TSs) requirements related to primary containment isolation instrumentation in accordance with the Nuclear Regulatory Commission-approved Technical Specification Task Force (TSTF), Improved Standard Technical Specifications change traveler, TSTF-306, Revision 2, “Add action to LCO 3.3.6.1 to give option to isolate the penetration.” The proposed amendment would revise TS Section 3.3.6.1, “Primary Containment Isolation Instrumentation,” by adding an ACTIONS note allowing intermittent opening, under administrative control, of penetration flow paths that are isolated. Additionally, the traversing in-core probe (TIP) system would be added as a separate isolation function with an associated Required Action to isolate the penetration within 24 hours rather than immediately initiating a unit shutdown.

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 addition of the note that the penetration flow path may be unisolated under administrative control simply provides consistency with what is already allowed elsewhere in TSs. The isolation function of the TIP valves is mitigative, and does not create any increased possibility of an accident. Also, the operation of the manual shear valves is unaffected by this activity. The ability to manually isolate the TIP system by either the normal isolation ball valves or the shear valves would be unaffected by the inoperable instrumentation. The Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated.

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

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

   Response: No

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. As a result no new failure modes are being introduced.

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

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

   Response: No

   The proposed change will not affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The allowance to unisolate a penetration flow path will not have a significant effect on the margin of safety because the penetration flow path can be isolated manually, if needed. This change simply provides consistency with what is already allowed elsewhere in TSs. The option to isolate a TIP penetration will ensure the penetration will perform as designed in the accident analysis. The ability to manually isolate the TIP system is unaffected by the inoperable instrumentation. The proposed change does not impact any safety analysis assumptions or results.

   Therefore, the proposed change does not result in 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: David T. Conley, Associate General Counsel II - Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, NC 27602.

NRC Branch Chief: Thomas H. Boyce.

Dominion Nuclear Connecticut Inc., et al., Docket No. 50-423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: November 23, 2009.

Description of amendment request: The proposed license amendment request would revise the Millstone Power Station, Unit 3 Technical Specification (TS) 6.8.4.g, “Steam Generator Program,” to exclude a portion of the tubes below the top of the steam generator tubesheet from periodic steam generator tube inspections. This request would also remove reference to the previous Cycle 13 interim alternate repair criteria.

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 previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed change to the steam generator tube inspection and repair criteria are the steam generator tube rupture (SGTR) event and the feedline break (FLB) postulated accidents.

During the SGTR event, the required structural integrity margins of the steam generator tubes and the tube.-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the tube-to-tubesheet joint. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded [pressurized-water reactor] PWR Steam Generator Tubes," are maintained for both normal and postulated accident conditions.

The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural integrity of the steam generator tubes and does not affect other systems, structures, components, or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to secondary leakage flow during the event. However, primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed changes since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter. Therefore, the proposed changes do not result in a significant increase in the consequences of a SGTR.

The consequences of a steam line break (SLB) are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a SLB) is limited by flow restrictions. These restrictions result from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications;

The leakage factor of 2.49 for Millstone Power Station Unit 3 (MPS3), for a postulated SLB/FLB, has been calculated as shown in Table RA124-2 of Enclosure 5. The leakage factor of 2.49 is a bounding value for all steam generators, both hot and cold legs, in Table RA124-2. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.49 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.49 and compared to the observed operational leakage.

The probability of a SLB is unaffected by the potential failure of a steam generator tube as the failure of the tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to- tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial and or circumferentially orientated cracks occurring 13.1 inches below the top of the tubesheet. The accident induced leak rate limit is 1.0 gpm. The technical specification (TS) operational leak rate is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.49 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

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

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

Response: No

The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident 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 change involve a significant reduction in a margin of safety?

Response: No

The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria maintains the required structural margins of the steam generator tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97-06, Revision 2, "Steam Generator Program Guidelines" and RG 1.121, are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the Nuclear Regulatory Commission (NRC) for meeting General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and GDC 32, "Inspection of Reactor Coolant Pressure Boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse Electric Company, LLC (Westinghouse) report WCAP-1 7071 -P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," defines a length of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

Therefore, the proposed change does not involve a significant reduction in any 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, Senior Counsel, Dominion Resource Services, Inc., 120 Tredegar Street, RS-2, Richmond, VA 23219

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: November 19, 2009.

Description of amendment request: The proposed change will correct identified non-conservatisms in Technical Specification 5.5.9 “Ventilation Filter Testing Program” by modifying the charcoal testing criteria to account for the 95% charcoal efficiency assumed for elemental iodine in the accident analyses for alternate source term.

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 testing acceptance criteria for the existing Indian Point 2 Control Room filtration system in Technical Specification (TS) 5.5.9 "Ventilation Filter Testing Program" to reflect current assumptions of iodine removal in accident dose calculations. The revised testing criteria does not add equipment or change the process for taking the test sample and only changes the test in the laboratory to be more restrictive. Therefore it cannot increase the probability of an accident occurring. The revised testing criteria is more stringent and therefore does not increase the consequences of an accident since it is more capable of mitigating control room doses and is consistent with existing 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 testing acceptance criteria for the existing Control Room filtration system. The proposed change does not involve installation of new equipment, modification of existing equipment, or result in a change to the way that the equipment or facility is operated so that no new equipment failure modes are introduced. Therefore the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   No. The proposed change revises the testing acceptance criteria for the existing Control Room filtration system. There is no change to the design requirements or the surveillance interval. The proposed change reflects the accident analysis dose calculation assumptions that assumed increased iodine removal. The factor of safety applied to the testing acceptance criteria remains the same. The new acceptance criterion is well within the system design capabilities. Therefore the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Nancy L. Salgado.

Entergy Nuclear Operations, Inc., Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3 (IP3), Westchester County, New York

Date of amendment request: December 15, 2009, as supplemented on December 22, 2009, January 4, 2010, and January 11, 2010.

<u>Description of amendment request</u>: The proposed amendment would allow a one-time extension of the 72-hour completion time of Technical Specification (TS) 3.7.5, Condition B, Action B.1 “Restore AFW [auxiliary feedwater] train to OPERABLE status” by 34 hours.

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

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

   No. The proposed change revises the allowed outage time (AOT) for the steam driven Auxiliary Boiler Feedwater Pump (ABFP) on a one time basis. Revising the AOT is not an accident initiator since an ABFP is a mitigating system. Therefore the proposed changes do not increase the probability of an accident occurring. The proposed AOT change is a one time increase that will allow repairs without the transient of shutdown. The plant is designed for single failure and recognizes that inoperability for short periods does not cause a significant increase in the consequences of an accident. The one time increase in this outage time is compensated with measures to reduce the potential need for the ABFP and the effects of events that could require the pump. Therefore the increase does not significantly increase the consequences of an accident. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   No. The proposed change revises the allowed outage time for the ABFP on a one time basis. The proposed change does not involve installation of new equipment or modification of existing equipment, so no new equipment failure modes are introduced. The proposed revision is not a change to the way that the equipment or facility is operated or analyzed and 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 reduction in the margin of safety associated with continued IP3 operation with Auxiliary Boiler Feedwater (ABF) pump 32 out of service during a 34 hour period beyond current allowed outage time is

> represented by an increase of approximately 50 percent in the allowed outage time. This change in the margin of safety has been compensated for by specific compensatory measures to reduce the potential need for the pump and to address postulated events that could require the pump. The increase in core damage frequency (CDF) associated with continued IP3 operation with ABFP 32 out of service for a duration of 106 hours which represents a 34 hour period beyond the current allowed outage time is 3.9E-5 per reactor year (ry). This results in an incremental conditional core damage probability (ICCDP) of 4.8E-07, which is below the ICCDP guidance threshold of 5E-07 identified in NRC Inspection Manual Part 9900. The ICCDP includes risk due to external events due to seismic, fire, and flood. The increase in large early release frequency (LERF) was estimated as 4.2E-7/ry (including external events), which results in an incremental conditional large early release probability (ICLERP) of 5.1E-9. Therefore the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Nancy L. Salgado.

Entergy Nuclear Operations, Inc., Docket Nos. 50-247 and 50-286, Indian Point Nuclear Generating Unit Nos. 2 and 3, Westchester County, New York

Date of amendment request: November 17, 2009.

Description of amendment request: The proposed change will correct identified non-conservatisms in the calculation of Emergency Diesel Generator (EDG) air receiver pressure requirements for Technical Specification (TS) 3.8.3. In addition, the proposed change will modify the number of normal EDG starts the air receiver is capable of providing as listed in the Final Safety Analysis Report.

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

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

   No. The proposed change revises the pressure at which the Emergency Diesel [G]enerator (EDG) air receiver is required to be kept to meet surveillance requirements, revises the minimum EDG air receiver pressure required for one start of the EDG, and changes the number of normal starts in the air receiver. Revising the air receiver upper and lower pressure limits and reducing the number of starts in the air receiver are not accident initiators since an EDG is a mitigating system. Therefore the proposed changes do not increase the probability of an accident occurring. The proposed changes will assure that each EDG is capable of starting consistent with assumed accident analyses. These analyses assume that an EDG starts the first time and accident analyses do not credit subsequent starts. The proposed new TS limits on the EDG air receiver will assure that air pressure is adequate to assure one attempt to start the EDG is available at the lower limit and will provide additional normal starts at the upper pressure established in the surveillance. Establishing acceptance criteria that replace non conservative criteria and assure the design bases is met assures the capability of equipment to mitigate accident conditions. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   No. The proposed change revises the pressure limit for the air receiver to initiate an alarm for low pressure, revises the lower pressure limit that must be maintained to assure that air is sufficient for at least one EDG start and revises the number of normal starts in the air receiver based on the revised calculations. The proposed change does not involve installation of new equipment or modification of existing equipment, so no new equipment failure modes are introduced. The proposed revision to the air receiver pressure limits and minimum air receiver EDG starts is also is [sic] not a change to the way that the equipment or facility is operated or analyzed and 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 surveillance tests, the conditions for failure of those tests and the number of EDG starts in the air receiver are means of assuring that the equipment is capable of maintaining the margin of safety established in the safety analyses for the facility. The proposed change in the EDG surveillance test acceptance criteria is consistent with values assumed in existing safety analyses which assume one start attempt for each EDG. The requirement for a minimum air pressure in the EDG air start receiver assures that there will be adequate air to allow at least one EDG start attempt which meets the intent of the existing TS. The reduction in the number of starts maintained in the air receiver does not affect the margins in accident analyses for this reason and because an EDG failure to start would reduce the air pressure below that required for one start before the overcrank timer would lock out a further start attempt. Therefore the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: November 23, 2009.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) 5.5.7, Inservice Testing Program, by replacing the references from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to the current code of record, the ASME Operation and Maintenance Nuclear Power Plants Code (ASME OM Code), the code of record for the James A. FitzPatrick Nuclear Power Plant (JAF)

Inservice Testing Program for Inservice Testing Program. This is an administrative amendment to maintain the TS current with the NRC accepted code of record for JAF.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed TS changes are non-technical, and are provided for consistency. There is no plant change involved, and thus, proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed TS changes are non-technical, i.e., there is no plant change involved, and thus, do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed TS changes are non-technical, i.e., there is no plant change involved. The changes are consistent with the regulations, and only update the TS to refer to the current code of reference. No design or safety margin is involved. Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

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

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

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: October 26, 2009.

Description of amendment request: The proposed change will revise Technical Specification (TS) 3.8.1 entitled "AC Sources - Operating" to extend, on a one-time basis, the allowable Completion Time (CT) of Required Action A.3 for one offsite circuit inoperable, from 72 hours to 14 days. This change is only applicable to startup transformer (ST) XST2 and will expire on March 1, 2011. This change is needed to allow sufficient time to make final terminations as part of a plant modification to facilitate connection of either ST XST2 or the spare ST to the Class 1E buses.

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 revise the CT for the loss of one offsite source from 72 hours to 14 days. The proposed one-time extension of the CT for the loss of one offsite power circuit does not significantly increase the probability of an accident previously evaluated. The startup transformers are not the initiator of any previously evaluated accidents involving a loss of offsite power (LOOP).

The TS will continue to require equipment that will power safety related equipment necessary to perform any required safety function. The one-time extension of the CT to 14 days does not affect the design of the STs, the interface of the STs with other plant systems, the operating characteristic of the STs, or the reliability of the STs.

Per Regulatory Guide (RG) 1.177, the risk acceptance guideline presented in RG 1.174 shows that Unit 1 met all the risk acceptance guidelines for delta core damage frequency (CDF), delta large early release frequency (LERF), incremental conditional core damage probability (ICCDP), and incremental conditional large early release probability (ICLERP). [CPSES,] Unit 2 met the same risk acceptance guidelines of delta LERF and ICLERP; however, the delta CDF and ICCDP were above the acceptance value. Since the increase above the regulatory guidance is small, and the risk reduction measures quantitatively addressed, the values for Unit 2 delta CDF and ICCDP would fall below the regulatory guidance as well as decrease the other risk metrics for both Units.

The consequence of a LOOP event has been evaluated in the CPNPP [Comanche Peak Steam Electric Station] Final Safety Analysis Report [ ] and the Station Blackout evaluation. Increasing the CT for one offsite power source on a one-time basis from 72 hours to 14 days does not increase the consequences of a LOOP event nor change the evaluation of LOOP events.

Therefore, the proposed change does 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 does not result in a change in the manner in which the electrical distribution subsystems provide plant protection. The proposed change will only affect the time allowed to restore the operability of the offsite power source through a startup transformer. The proposed change does not affect the configuration, or operation of the plant. The proposed change to the CT will facilitate installation of a plant modification which will improve plant design and will eliminate the necessity to shut down both Units if [ST] XST2 fails or requires maintenance that goes beyond the current TS CT of 72 hours. This change will improve the long-term reliability of the 345kV [kiloVolt] offsite circuit STs which are common to both CPNPP Units.

There are no changes to the STs or the supporting systems operating characteristics or conditions. The change to the CT does not change any existing accident scenarios, nor create any new or different accident scenarios. In addition, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter any of the assumptions made in the safety analysis.

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

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

Response: No.

The proposed change does not affect the acceptance criteria for any analyzed event nor is there a change to any safety limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Neither the safety analyses nor the safety analysis acceptance criteria are affected by this change. The proposed change will not result in plant operation in a configuration outside the current design basis. The proposed activity only increases, for a one-time pre-planned occurrence, the period when the plant may operate with one offsite power source. The margin of safety is maintained by maintaining the ability to safely shut down the plant and remove residual heat.

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.

PSEG Nuclear LLC, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: November 4, 2009.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to: (1) delete TS 4.0.5, which pertains to surveillance requirements (SRs) for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable; and (4) revise TS 6.10.3.h to reflect the deletion of the ISI Program from the TSs. The new TS for the IST Program, TS 6.8.4.i, will indicate that the program will include testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), replacing the current reference to Section XI of the ASME Code specified in TS 4.0.5. In addition, TS 6.8.4.i would revise the requirements, currently contained in TS 4.0.5, regarding the applicability of the surveillance interval extension provisions of SR 4.0.2.

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 revise TS 4.0.5, Surveillance Requirements for Inservice Inspections and Testing of ASME Code Components, for consistency with 10 CFR 50.55a(f)(4) requirements regarding inservice testing of pumps and valves. The proposed change incorporates revisions to the ASME OM Code and clarifies testing frequency requirements for testing pumps and valves. The proposed change also relocates the ISI and IST Programs consistent with NUREG-1433. A commitment is made to maintain [Generic Letter (GL)] 88-01 inspection requirements in the ISI Program.

   The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility.

Therefore, the proposed changes do not represent 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 changes do not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

Response: No

The proposed changes revises and relocates TS 4.0.5, Surveillance Requirements for Inservice Inspections and Testing of ASME Code Components, for consistency with (1) the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves and (2) NUREG-1433. The proposed change updates references to the ASME OM Code, clarifies testing frequency requirements for testing pumps and valves, and relocates the IST Program to Section 6.0 of TS, and the ISI Program to a licensee controlled document. The safety function of the affected pumps and valves will be maintained; the programs will continue to be implemented with the required regulations and codes. A commitment is made to maintain GL 88-01 inspection requirements in the ISI Program; there will be no change to these requirements.

Therefore, this 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: Vincent Zabielski, PSEG Nuclear LLC - N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Harold K. Chernoff.

PSEG Nuclear LLC, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: December 1, 2009.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to change the required frequency of testing control rod scram times from "at least once per 120 days of POWER OPERATION" to "at least once per 200 days of POWER OPERATION." This change is based on TS Task Force (TSTF) change traveler TSTF-460, Revision 0, "Control Rod Scram Time Testing Frequency." TSTF-460 has been approved generically by the Nuclear Regulatory Commission (NRC) for incorporation into the boiling water reactor (BWR) Standard TS (STS); NUREG-1433 (BWR/4) and NUREG-1434 (BWR/6). The NRC staff published a notice announcing the availability of this proposed TS change using the consolidated line item improvement process (CLIIP) in the *Federal Register* on August 23, 2004 (69 FR 51864). Since Hope Creek Generating Station has not adopted the STS, the licensee has proposed variations from the CLIIP to ensure consistency with NUREG-1433, Revision 3, "Standard Technical Specifications, General Electric Plants, BWR/4." The changes to align with NUREG-1433 involve the adoption of a revised control rod scram time test methodology and an establishment of a category of operable but "slow" control rods.

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 changes extend the frequency and revise the evaluation methodology for control rod scram times, and identify a new category of "slow" control rods for assessing control rod operability. The frequency of control rod scram 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, because the tested component is still required to be operable. The proposed evaluation methodology is consistent with industry approved methods and ensures control rod operability requirements for the number and distribution of operable, slow, and stuck control rods [and] continue[s] to satisfy scram reactivity rate assumptions used in plant safety analysis. 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 changes do not involve any physical alteration of the plant (no new or different type of equipment is being installed) and do not involve a change in the design, normal configuration, or basic operation of the plant. The proposed changes do not introduce any new accident initiators. The proposed changes do not involve significant changes in the fundamental methods governing normal plant operation and do not require unusual or uncommon operator actions. The proposed changes provide assurance that the plant will not be operated in a mode or condition that violates the assumptions or initial conditions in the plant safety analyses and that [structures, systems and components] remain capable of performing their intended safety functions as assumed in the same analyses. Consequently, the response of the plant and the plant operator to postulated events will not be significantly different. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

   Margin of safety is related to confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. The proposed changes address control rod scram test

> performance and acceptance criteria as well as control rod operability requirements. The scram test acceptance criteria and control rod operability restrictions are based on industry approved methodology and will continue to ensure control rod scram design functions and reactivity insertion assumptions used in plant safety analyses continue to be protected. The proposed changes also extend the frequency of testing control rod scram times while at-power from 120 days to 200 days. The proposed change continues to test the control rod scram time to ensure the assumptions in the plant safety analysis are protected. The demonstrated reliability of the control rod scram function justifies the extension of the surveillance frequency. Therefore, the proposed 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: Vincent Zabielski, PSEG Nuclear LLC - N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Harold K. Chernoff.

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: December 16, 2009.

Description of amendment request: The amendment would revise the Technical Specifications (TS) to adopt Nuclear Regulatory Commission (NRC)-approved Revision 2 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-427, "Allowance for Non Technical Specification Barrier Degradation on Support System Operability." The proposed amendment would modify the requirements for unavailable barriers by adding Limiting Condition for Operation 3.0.9.

The NRC staff published a notice of opportunity for comment in the *Federal Register* on June 2, 2006 (71 FR 32145), on possible amendments adopting TSTF-427, including a model safety evaluation and model no significant hazards consideration (NSHC) Determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the *Federal Register* October 3, 2006 (71 FR 58444). The licensee affirmed the applicability of the following NSHC determination in its application dated December 16, 2009.

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

> Criterion 1--The Proposed Change Does Not Involve a Significant Increase in the Probability of Consequences of an Accident Previously Evaluated
>
> The proposed change allows a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an unavailable hazard barrier if risk is assessed and managed. The postulated initiating events which may require a functional barrier are limited to those with low frequencies of occurrence, and the overall TS system safety function would still be available for the majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on the allowance provided by proposed LCO 3.0.9 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.9. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, 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 Previously Evaluated
>
> The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Allowing delay times for entering supported system TS when inoperability is due solely to an unavailable hazard barrier, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an unavailable barrier, if risk is assessed and managed. The postulated initiating events which may require a functional barrier are limited to those with low frequencies of occurrence, and the overall TS system safety function would still be available for the majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.9 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The net change to the margin of safety is insignificant as indicated by the anticipated low levels of associated risk (ICCDP [incremental conditional core damage probability] and ICLERP [incremental conditional large early release probability]) as shown in Table 1 of Section 3.1.1 in the Safety Evaluation [published in the *Federal Register* on October 3, 2006 (71 FR 58444)]. Therefore, this change does not involve a significant reduction in a margin of safety.

Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar Street, RS-2, Richmond, VA 23219.

NRC Branch Chief: Gloria Kulesa.

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

Date of amendment request: October 10, 2009.

Description of amendment request: The proposed changes will revise Technical Specification (TS) 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N\Delta H$), TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," for use of the Best Estimate Analyzer for Core Operations - Nuclear

(BEACON) Power Distribution Monitoring System (PDMS) described in WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," to perform power distribution surveillances.

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

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

   Response: No.

   The PDMS performs continuous core power distribution monitoring with data input from existing plant instrumentation. This system utilizes an NRC [U.S. Nuclear Regulatory Commission] approved Westinghouse proprietary computer code, i.e., Best Estimate Analyzer for Core Operations - Nuclear (BEACON), to provide data reduction for incore flux maps, core parameter analysis, load follow operation simulation, and core prediction. The PDMS does not provide any protection or control system function. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Safety Analysis Report (USAR).

   Use of the PDMS supports maintaining the core power distribution within required limits. Further continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (i.e., earlier determination of an adverse condition developing) for operator action prior to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

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

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

Response: No.

Other than use of the PDMS to monitor core power distribution, implementation of the PDMS and associated Technical Specification changes has no impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with implementation of the PDMS do not result in a change to the design basis of any plant component or system. The evaluation of the effects of using the PDMS to monitor core power distribution parameters shows that all design standards and applicable safety criteria limits are met.

The proposed changes do not result in any event previously deemed incredible being made credible. Implementation of the PDMS will not result in any additional adverse condition and will not result in any increase in the challenges to safety systems. The cycle-specific variables required by the PDMS are calculated using NRC-approved methods. The Technical Specifications will continue to require operation within the required core operating limits, and appropriate actions will continue to be taken when or if limits are exceeded.

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

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

Response: No.

No margin of safety is adversely affected by the implementation of the PDMS. The margins of safety provided by current Technical Specification requirements and limits remain unchanged, as the Technical Specifications will continue to require operation within the core limits that are based on NRC-approved reload design methodologies. Appropriate measures exist to control the values of these cycle-specific limits, and appropriate actions will continue to be specified and taken for when limits are violated. Such actions remain unchanged.

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

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

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: November 20, 2009.

Description of amendment request: The proposed changes will revise Technical Specification (TS) 3.8.1, "AC [Alternating Current] Sources - Operating," by adding a Note to the Required Actions B.3.1 and B.3.2 to indicate that the TS 3.8.1 Required Actions B.3.1 and B.3.2 are satisfied if the diesel generator (DG) became inoperable due to an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing. The amendment also proposes to revise the Completion Times for Required Actions B.3.1 and B.3.2 to specify a Completion Time based on the discovery of an issue or failure of the DG.

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.

   WCNOC [Wolf Creek Nuclear Operating Corporation] is proposing to add a Note to Required Actions B.3.1 and B.3.2 to indicate that the TS 3.8.1

Required Actions of B.3 are satisfied if the DG became inoperable due to an inoperable support system, an independently testable component or preplanned preventative maintenance or testing. The proposed change to the TS does not involve a change in the operational limits or physical design of the emergency power system. Diesel generator (DG) OPERABILITY and reliability will continue to be assured while minimizing the potential number of required DG starts. The DGs are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased.

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

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

   Response: No.

   No new or different accidents result for implementing the proposed change. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The change does not alter assumptions made in the safety analysis for DG performance.

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

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

   Response: No.

   The proposed 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 impacted by this change. The proposed change will not result in operation in a configuration outside the design basis.

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

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

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

NRC Branch Chief: Michael T. Markley.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

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

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the *Federal Register* as indicated.

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or

Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr.resource@nrc.gov.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: August 26, 2009.

Brief description of amendment: The proposed amendment would revise the Technical Specification (TS) Section 6.5 that governs administrative controls of High Radiation Areas (HRA) to incorporate the HRA administrative controls contained within the Standard Technical Specifications, NUREG-1433, Revision 3.

Date of Issuance: January 4, 2010.

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

Amendment No.: 241.

Facility Operating License No. DPR-28: Amendment revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 20, 2009 (74 FR 53778).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 4, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 22, 2009.

Brief description of amendments: The amendments revise the Technical Specification (TS) definition of the fully withdrawn position of the Rod Cluster Control Assemblies (RCCAs) to minimize localized RCCA wear. Previously, the fully withdrawn position for the RCCAs was defined in the TSs as being within the interval of 222 to 228 steps withdrawn (i.e., steps above rod bottom). The approved change allows the fully withdrawn position to be defined as being within the interval of 222 to 230 steps withdrawn.

Date of issuance: January 12, 2010.

Effective date: As of the date of issuance. The Salem Unit No. 1 amendment shall be implemented prior to entering Mode 2 following refueling outage 1R20 (currently scheduled for spring 2010). The Salem Unit No. 2 amendment shall be implemented prior to entering Mode 2 following refueling outage 2R18 (currently scheduled for spring 2011).

Amendment Nos.: 292 and 276.

Facility Operating License Nos. DPR-70 and DPR-75: The amendments revised the TSs and the License.

Date of initial notice in FEDERAL REGISTER: June 2, 2009 (74 FR 26435).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 12, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 13th day of January 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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