NUCLEAR REGULATORY COMMISSION

[NRC–2009–0518]

Biweekly Notice Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 5, 2009, to November 18, 2009. The last biweekly notice was published on November 17, 2009 (74 FR 59259).

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 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, 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 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 petitioner/requestor should contact the Office of the Secretary by e-mail at *hearing.docket@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 using the agency's adjudicatory e-filing system 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 Meta-System Help Desk, which is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays. The Meta-System 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 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.

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 request and/or petition 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 submissions.

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: September 28, 2009.

Description of amendment request: The amendments would revise Required Action A.1 of Technical Specification (TS) 3.8.7, ''Inverters—Operating,'' for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, by extending the Completion Time for restoration of an inoperable vital alternating current (AC) inverter from 24 hours to 7 days.

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 TS amendment does not affect the design of the vital AC inverters, the operational characteristics or function of the inverters, the interfaces between the inverters and other plant systems, or the reliability of the inverters. An inoperable vital AC inverter is not considered an initiator of an analyzed event. In addition, Required Actions and the associated Completion Times are not initiators of previously evaluated accidents. Extending the Completion Time for an inoperable vital AC inverter would not have a significant impact on the frequency of occurrence of an accident previously evaluated. The proposed amendment will not result in modifications to plant activities associated with inverter maintenance, but rather, provides operational flexibility by allowing additional time to perform inverter troubleshooting, corrective maintenance, and post-maintenance testing on-line.

The proposed extension of the Completion Time for an inoperable vital AC inverter will not significantly affect the capability of the inverters to perform their safety function, which is to ensure an uninterruptible supply of 120-volt AC electrical power to the associated power distribution subsystems. An evaluation, using PRA [probabilistic risk assessment] methods, confirmed that the increase in plant risk associated with implementation of the proposed Completion Time extension is consistent with the NRC's Safety Goal Policy Statement, as further described in [NRC Regulatory Guide] RG 1.174 and RG 1.177. In addition, a deterministic evaluation concluded that plant defense-in-depth philosophy will be maintained with the proposed Completion Time extension. Based on the above, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment does not involve physical alteration of the PVNGS. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the PVNGS is operated. There are no setpoints at which protective or mitigating actions are initiated that are affected by this proposed action. The use of the alternate Class 1E power source for the vital AC instrument bus is consistent with the PVNGS plant design. The change does not alter assumptions made in the safety analysis. This proposed action will not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No alteration is proposed to the procedures that ensure the PVNGS remains within analyzed limits, and no change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

Based on the above, the proposed amendment 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.

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms or actions. The proposed amendment does not alter the design or configuration of the vital AC inverters or their associated 120-volt AC subsystems, and does not alter the setpoints at which alarms and associated actions are initiated. With one of the required 120-volt AC vital instrumentation buses being powered from the alternate safety-related Class 1E power supply, which is backed by the divisional diesel generator (DG), there is no significant reduction in the margin of safety. Testing of the DGs and associated electrical distribution equipment provides confidence that the DGs will start and provide power to the associated equipment in the unlikely event of a loss of offsite power during the extended 7-day Completion Time.

Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, sufficient safety margins will be maintained, and any increases in risk are consistent with the NRC Safety Goal Policy Statement. Furthermore, during the proposed extended inverter Completion Time, any increases in risk posed by potential combinations of equipment out of service will be managed in accordance with the PVNGS site Configuration Risk Management Program, consistent with Paragraph (a)(4) of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants.''

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

The NRC staff has reviewed the licensee's analysis and, based on 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: August 18, 2009.

Description of amendments request: The proposed license amendments revise Technical Specification 3.3.1.1, ''Reactor Protection System (RPS) Instrumentation,'' Surveillance Requirement 3.3.1.1.8, to increase the frequency interval between local power range monitor calibrations from 1100 megawatt-days per metric ton average core exposure (i.e., equivalent to approximately 907 effective full-power hours (EFPH)) to 2000 EFPH.

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 amendments revise the surveillance interval for the LPRM [local power range monitor] calibration from 1100 MWD/T [megawatt days per metric ton] average core exposure to 2000 effective full power hours (EFPH). Increasing the frequency interval between required LPRM calibrations is acceptable due to improvements in fuel analytical bases, core monitoring processes, and nuclear instrumentation. The revised surveillance interval continues to ensure that the LPRM detector signal will continue to be adequately calibrated.

This change will not alter the operation of process variables, structures, systems, or components as described in the Updated Final Safety Analysis Report. The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed change does not alter the initiation conditions or operational parameters for the LPRM subsystem and there is no new equipment introduced by the extension of the LPRM calibration interval. The performance of the Average Power Range Monitor (APRM), Rod Block Monitor (RBM), and Oscillation Power Range Monitor (OPRM) systems is not affected by the proposed surveillance interval increase. The proposed LPRM calibration interval extension will have no significant effect on the Reactor Protection System (RPS) instrumentation accuracy during power maneuvers or transients and will, therefore, not significantly affect the performance of the RPS. As such, no individual precursors of an accident are affected and the proposed amendments do not increase the probability of a previously analyzed event.

The radiological consequences of an accident can be affected by the thermal limits existing at the time of the postulated accident; however, increasing the surveillance interval frequency will not increase the calculated thermal limits since all uncertainties associated with the increased interval are currently implemented and are currently used to calculate the existing safety limits. Plant specific evaluation of LPRM sensitivity to exposure has determined that the extended calibration frequency increases the LPRM signal uncertainty value used in the SLMCPR [safety limit for minimum critical power] analysis; however, the increase is bounded by the values currently used in the safety analysis. Therefore, the thermal limit calculation is not significantly affected by LPRM calibration frequency, and thus the radiological consequences of any accident previously evaluated are not increased.

Based on the above, the proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The performance of the APRM, RBM, and OPRM systems are not affected by the proposed LPRM surveillance interval increase. The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. For the proposed LPRM extended calibration interval frequency, all uncertainties remain less than the uncertainties assumed in the existing thermal limit calculations. The proposed change does not change or introduce any new equipment, modes of system operation, or failure mechanisms; therefore, no new accident precursors are created. Based on the above information, the proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change has no impact on equipment design or fundamental operation, and there are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed LPRM surveillance interval increase. The performance of the APRM, RBM, and OPRM systems are not affected by the proposed change. The margin of safety can be affected by the thermal limits existing at the time of the postulated accident; however, uncertainties associated with LPRM chamber exposure have no significant effect on the calculated thermal limits. Plant-specific evaluation of LPRM sensitivity to exposure has determined that the extended calibration frequency increases the LPRM signal uncertainty value used in the SLMCPR analysis; however, the increase is bounded by the values currently used in the safety analysis. The thermal limit calculation is not significantly affected since LPRM sensitivity with exposure is well defined. LPRM accuracy remains within that used to determine the total power uncertainty assumed in the thermal analysis basis, therefore maintaining thermal limits and the safety margin. The proposed change does not affect uncertainties or initial conditions assumed in the thermal limit calculations and therefore the margin of safety in the safety analyses is maintained. Based on the above information, the proposed amendments do 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.

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

Date of amendment request: October 19, 2009.

Description of amendment request: The proposed amendment relocates the Waterford Steam Electric Station, Unit 3 Steam Generator Level—High trip requirements from Technical Specification Sections 2.2 and 3/4.3.1 to the Technical Requirements Manual.

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 relocates the Steam Generator Level—High Trip to a licensee-controlled document. The Steam Generator (SG) Level—High trip function is not credited in any DBA [design-basis accident] or transient analysis and is not an initiator to any accident analysis. As a result, neither the probability nor the consequences of an accident previously evaluated are significantly increased by this change.

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

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

Response: No.

The proposed change relocates the Steam Generator Level—High trip function to a licensee-controlled document. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

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

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

Response: No.

The proposed change relocates the Steam Generator Level—High trip function to a licensee-controlled document. This will allow changes to the Steam Generator Level—High Trip requirements currently in the Technical Specifications to be performed in accordance with the requirements of 10 CFR 50.59. As the Steam Generator Level—High trip function has been determined to not meet the definition of Technical Specifications or the criteria in 10 CFR 50.36 (c)(2)(ii), lack of NRC review and approval prior to implementation for changes that are not determined to be a significant hazard will not lead to a significant reduction in the margin of safety.

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

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

Attorney for licensee: Terence A. Burke, Associate General Counsel—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: September 24, 2009.

Description of amendment request: The amendment request proposes a one-time extension of the Completion Time (CT) to restore a unit-specific essential service water train to operable status associated with Technical Specification Limiting Condition for Operation (LCO) 3.7.8, Essential Service Water (SX) System, from 72 hours to 144 hours. The proposed change will only be used one time during the Byron Station Unit 2 spring 2010 refueling outage. The licensee is requesting an extension of the CT to 144 hours to replace two of the four SX pump suction isolation valves; maintenance history has shown that replacement of the SX pump suction isolation valves cannot be assured within the existing 72 hour CT window.

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 have been evaluated using the risk-informed processes described in Regulatory Guide (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' dated July 1998 and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' dated August 1998. In addition, proposed revised guidance as described in Draft Regulatory Guide DG–1226, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' and Draft Regulatory Guide DG–1227, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' was reviewed for insights. The risk associated with the proposed changes was shown to be acceptable.

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The SX system is not considered an initiator for any of these previously analyzed events. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. No active or passive failure mechanisms that could lead to an accident are affected. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The unit-specific SX system consists of two separate, electrically independent, 100% capacity, safety related, cooling water trains. Each train consists of a 100% capacity pump, piping, valving, and instrumentation. Normally, the pumps and valves are remotely and manually aligned. However, the pumps are automatically started upon receipt of a safety injection signal or an undervoltage on the engineered safety features (ESF) bus, and all essential valves are aligned to their post accident positions. The SX system is also the backup water supply to the auxiliary feedwater system and fire protection system.

The design basis of the SX system is for one SX train, in conjunction with the component cooling water (CC) system and a 100% capacity containment cooling system, to remove core decay heat following a design basis LOCA [loss-of-coolant accident] as discussed in the UFSAR [updated final safety analysis report], Section 6.2, ''Containment Systems.'' This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the reactor coolant system by the emergency core cooling system pumps. The SX system is designed to perform its function with a single failure of any active component, assuming the loss of offsite power. The proposed one-time increase in the CT is consistent with the philosophy of the current Technical Specification LCO which allows one train of SX to be inoperable for 72 hours. This change only extends the 72 hour Completion Time to 144 hours which has been shown to be acceptable from a risk perspective; therefore, the proposed change does not involve a significant increase in the 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 the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. Based on this evaluation, 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 does not alter any existing setpoints at which protective actions are initiated and no new setpoints or protective actions are introduced. The design and operation of the SX system remains unchanged. The risk associated with the proposed increase in the time an SX pump is allowed to be inoperable was evaluated using the risk-informed processes described in RG 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' dated July 1998 and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' dated August 1998. The risk was shown to be acceptable. Based on this evaluation, 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: Stephen J. Campbell.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Beaver County, Pennsylvania

Date of amendment request: July 6, 2009.

Description of amendment request: The proposed amendment would revise Technical Specification 5.6.3, ''Core Operating Limits Report,'' to allow the use of the generically approved Topical Report, WCAP–16009–P–A, ''Realistic Large Break LOCA [Loss-of-Coolant Accident] Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),'' for BVPS–1.

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. No physical changes are required as a result of implementing the ASTRUM best-estimate large break [LOCA] methodology and associated technical specification changes. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph (b). No other accident is potentially affected by this change.

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

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

No. There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

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

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

No. The methodology used in the analysis would more realistically describe the expected behavior of plant systems during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties are analyzed to provide assurance that the most severe postulated LOCAs are calculated. As described in Section 3.3, there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph (b) are met.

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: David W. Jenkins, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: September 9, 2009.

Description of amendment request: The proposed amendment would change the frequency of control rod notch testing, as specified in Technical Specification (TS) surveillance requirement (SR) 4.1.3.1.2.a, from at least once per 7 days to at least once per 31 days. The purpose of this SR is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. This ensures that the control rod is not stuck and is free to insert on a scram signal. The proposed amendment would also add the word ''fully'' to the Action for TS Limiting Condition for Operation (LCO) 3.9.2 to clarify the requirement to fully insert all insertable control rods when the required source range monitor (SRM) instrumentation is inoperable. The licensee stated that the proposed amendment is based on Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) change, TSTF–475, Revision 1, ''Control Rod Notch Testing Frequency and SRM Insert Control Rod Action.'' The availability of this change to the Standard Technical Specifications (STS) was announced in the **Federal Register** on November 13, 2007 (72 FR 63935) as part of the consolidated line item improvement process. The **Federal Register** notice included a model safety evaluation, a model application and a model proposed a no significant hazards consideration (NSHC) determination. In its application dated September 9, 2009, the licensee affirmed the applicability of the proposed NSHC determination for TSTF–475 and has incorporated it by reference to satisfy the requirements of 10 CFR 50.91(a). Since Hope Creek Generating Station has not adopted the STS (*e.g.,* NUREG–1433), the licensee has proposed minor variations from the TS changes described in TSTF–475.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff's review is presented below.

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

Response: No.

The proposed change to SR 4.1.3.1.2.a reduces the frequency of control rod notch testing. Changing the frequency of testing is not expected to have any significant impact on the reliability of the control rods to insert as required on a scram signal. The proposed change to the Action for LCO 3.9.2 merely clarifies the intent of the action. There are no physical plant modifications associated with this change. The proposed amendment would not alter the way any structure, system, or component (SSC) functions and would not alter the way the plant is operated. As such, the proposed amendment would have no impact on the ability of the affected SSCs to either preclude or mitigate 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 proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment would not change the design function or operation of the SSCs involved and would not impact the way the plant is operated. As such, the proposed change would not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. 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 margin of safety is associated with the confidence in the ability of the fission product barriers (*i.e.,* fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation to the public. There are no physical plant modifications associated with the proposed amendment. The proposed amendment would not alter the way any SSC functions and would not alter the way the plant is operated. The proposed amendment would not introduce any new uncertainties or change any existing uncertainties associated with any safety limit. The proposed amendment would have no impact on the structural integrity of the fuel cladding, reactor coolant pressure boundary, or containment structure. Based on the above considerations, the NRC staff concludes that the proposed amendment would not degrade the confidence in the ability of the fission product barriers to limit the level of radiation to the public. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: October 20, 2009.

Description of amendment request: The proposed amendment would delete paragraph d of Technical Specification 5.2.2, ''Unit Staff,'' superseded by Title 10 of the Code of Federal Regulations Part 26, Subpart I.

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

The proposed change removes Technical Specification (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 Part 26. 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.

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 Part 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 affect 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.

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 Part 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 shut down the plant and to maintain the plant in a safe shut down condition.

Removal of plant-specific TS administrative requirements will not reduce a margin of safety because the requirements in 10 CFR Part 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas Boyce.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: October 20, 2009.

Description of amendment request: The proposed amendment would delete paragraph g of Technical Specification 6.2.2, ''Facility Staff,'' which was superseded by Title 10 of the Code of Federal Regulations (10 CFR), Part 26, Subpart I. This change is consistent with Nuclear Regulatory Commission approved Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler TSTF–511, Revision 0, ''Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26.''

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

The proposed change removes Technical Specification (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 Part 26. 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.

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 Part 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 affect 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.

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 Part 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 shut down 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 Part 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas H. Boyce.

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: September 28, 2009.

Description of amendment request: The proposed changes would address the filtration function of the Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS) and are consistent with the associated design and licensing basis accident analysis assumptions. The proposed changes will add new Conditions B and C with associated Action Statements and Completion Times to Technical Specification (TS) 3.7.12 and modify Conditions A and D.

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 license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes do not adversely affect accident initiators or precursors and do not alter the design assumptions, conditions, or configuration of the facility. The new conditions only affect the filtration function of ECCS PREACS, which is an accident mitigation function, so accident initiation probability is not impacted. Regarding significance of the proposed changes relative to the accident consequences, the new conditions remain consistent with existing design assumptions (i.e., dose calculations show that the filtration function is not required when ECCS leakage is less than the maximum allowable unfiltered leakage) and filtration is required to be operable as required to support the design analysis assumptions.

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

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

The addition of the new Conditions B and C with associated Action Statements and Completion Times to TS 3.7.12 and modification of Condition D to address the filtration function of ECCS PREACS does not impact the accident analysis or associated assumptions. The new conditions only address actions to be taken when portions of ECCS PREACS (an accident mitigation system) is out-of-service.

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?

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The proposed new conditions recognize that there may be limited leakage situations when filtration is not required to meet the accident analysis assumptions. Allowing safety equipment to be inoperable while it is not required is not reducing the analyzed 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 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, Virginia 23219.

NRC Branch Chief: Gloria J. Kulesa.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: October 16, 2009.

Description of amendment request: The license amendment request (LAR) adds two references to the list of NRC approved methodologies contained in the Technical Specifications (TSs). Specifically, Westinghouse document WCAP–8745–P–A, ''Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function,'' and the Dominion Fleet Report DOM–NAF–2–A, ''Reactor Core Thermal-Hydraulics Using the VIPRE–D Computer Code,'' including Appendix B, ''Qualification of the Westinghouse WRB–1 CHF [Critical Heat Flux] Correlation in the Dominion VIPRE–D Computer Code,'' in TS 6.2.C as a referenced analytical methodology report.

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

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

Response: No.

Approval of the proposed changes will allow Dominion to use the VIPRE–D/WRB–1 and VIPRE–D/W–3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores, using the DDLs [Deterministic Design Limits] documented in Appendix B of the DOM–NAF–2–A Fleet Report and the SDL [Statistical Design Limit]. Neither the code/correlation pair nor the Statistical Departure from Nucleate Boiling Ratio (DNBR) Evaluation Methodology make any contribution to the potential accident initiators and thus cannot increase the probability of any accident. Further, since both the deterministic and statistical DNBR limits meet the required design basis of avoiding Departure from Nucleate Boiling (DNB) with 95% probability at a 95% confidence level, the use of the new code/correlation and the Statistical DNBR Evaluation Methodology do not increase the potential consequences of any accident. Finally, the full core DNB design limit provides increased assurance that the consequences of a postulated accident which includes radioactive release would be minimized because the overall number of rods in DNB would not exceed the 0.1% level. The pertinent evaluations to be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable have been performed and determined to be acceptable. The use of a different code/correlation pair will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. The use of the VIPRE–D/WRB–1 and VIPRE–D/W–3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores will not result in a measurable impact on normal operating plant releases and will not increase the predicted radiological consequences of accidents postulated in the UFSAR [Updated Final Safety Analysis Report].

The remaining proposed changes are being made to enhance the completeness of the Surry TS and to achieve consistency with NUREG–1431 Rev. 3. The proposed changes do not add or modify any plant systems, structures or components (SSCs). The proposed changes to relocate TS parameters to the COLR [Core Operating Limits Report] are programmatic and administrative in nature. These changes do not physically alter safety-related systems nor affect the way in which safety-related systems perform their functions. Additional Safety Limits on the DNB design basis and peak fuel centerline temperature are being imposed in TS 2.1, ''Safety Limit, Reactor Core,'' and the Reactor

Core Safety Limits figure is being relocated to the COLR. The additional Safety Limits are consistent with the values stated in the UFSAR and those being proposed herein. The proposed changes do not, by themselves, alter any of the relocated parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. TS 6.2.C continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies and that applicable limits of the safety analyses are met. Deletion of the obsolete limits associated with N–1 loop operation (TS 2.1.A.2, TS 2.1.A.3, TS Figure 2.1–2, TS Figure 2.1–3) and fuel densification (TS figure 2.1–4) is acceptable since these limits no longer represent limiting conditions for operation and are not required to be in the Technical Specifications.

Thus, the proposed changes do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this 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 changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed).

The use of VIPRE–D and its applicable fuel design limits for DNBR does not impact any of the applicable design criteria and all pertinent licensing basis criteria will continue to be met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Setpoint safety analysis evaluations have demonstrated that the use of VIPRE–D is acceptable. Design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the VIPRE–D code/correlation or the Statistical DNBR Evaluation Methodology does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

The proposed change adds a new surveillance requirement of RCS [Reactor Coolant System] Total Flow Rate and requests the addition of an already approved method for determining plant operating limits. The proposed change does not adversely affect accident initiators or precursors, nor does it alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits.

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

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

Response: No.

The proposed changes to relocate TS parameters to the COLR are programmatic and administrative in nature. Additional Safety Limits on the DNB design basis and peak fuel centerline temperature are being imposed in TS 2.1, ''Safety Limit, Reactor Core,'' and the Reactor Core Safety Limits figure is being relocated to the COLR. The additional Safety Limits are consistent with the values stated in the UFSAR and those being proposed herein.

Approval of the proposed changes will allow Dominion to use the VIPRE–D/WRB–1 and VIPRE–D/W–3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores, using the DDLs documented in Appendix B of the DOM–NAF–2–A Fleet Report and the SDL documented herein. The SDL has been developed in accordance with the Statistical DNBR Evaluation Methodology. The DNBR limits meet the design basis of avoiding DNB with 95% probability at a 95% confidence level. The use of the VIPRE–D/WRB–1 code/correlation provides the same margin to safety as the current code/correlation COBRA/WRB–1 used at Surry.

Therefore, the proposed TS 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, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar St., RS–2, Richmond, VA 23219.

NRC Branch Chief: Gloria Kulesa.

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

FPL Energy Duane Arnold, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: April 17, 2009.

Brief description of amendment: The amendment revises Operating License No. DPR–49 by changing ''FPL Energy Duane Arnold, LLC'' to ''NextEra Energy Duane Arnold, LLC,'' where appropriate, to reflect the renaming of FPL Energy Duane Arnold, LLC to NextEra Energy Duane Arnold, LLC.

Date of issuance: November 13, 2009.

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

Amendment No.: 275.

Facility Operating License No. DPR–49: The amendment revised the License and Appendix B—Additional Conditions.

*Date of initial notice in **Federal Register**:* June 30, 2009 (74 FR 31324).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 2009.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: June 2, 2009.

Brief description of amendment: The amendment (1) deleted Technical Specification (TS) surveillance requirement (SR) 3.1.3.2 and revised SR 3.1.3.3, (2) removed reference to SR 3.1.3.2 from Required Action A.3 of TS 3.1.3, ''Control Rod OPERABILITY,'' and (3) revised Example 1.4–3 in Section 1.4, ''Frequency,'' to clarify the applicability of the 1.25 surveillance test interval extension. The changes are in accordance with NRC-approved TS Task Force (TSTF) traveler TSTF–475, Revision 1, ''Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action.''

Date of issuance: November 12, 2009.

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

Amendment No.: 235.

Facility Operating License No. DPR–46: Amendment revised the Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register:*** June 30, 2009 (74 FR 31325).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 2009.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, (SSES Units 1 and 2) Luzerne County, Pennsylvania

Date of application for amendments: March 24, 2009, as supplemented by letters dated April 24, and September 11, 2009.

Brief description of amendments: The change revised the allowable value in the Technical Specification (TS) Table 3.3.5.1–1 (Function 3.d) for the high-pressure coolant injection automatic pump suction transfer from the condensate storage tank (CST) to the suppression pool. The present allowable value for this transfer is greater than or equal to 36 inches above the CST bottom. The change is to increase the allowable value for this transfer to occur at greater than or equal to 40.5 inches above the CST bottom.

Additionally, the amendment also included an editorial/administrative change which corrected a typographical error in the SSES Units 1 and 2 TS Section 3.10.8.f.

Date of issuance: November 9, 2009.

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

Amendment Nos.: 254 for Unit 1 and 234 for Unit 2.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the License and Technical Specifications.

*Date of initial notice in **Federal Register:*** October 6, 2009, (74 FR 51332).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 9, 2009.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: July 28, 2009, supplemented by letters dated September 16 and 30, 2009.

Brief Description of amendments: These amendments revise the Technical Specifications (TS) of Surry Power Station, Units 1 and 2. The request proposed changes to the inspection scope and repair requirements of TS Section 6.4.Q, ''Steam Generator (SG) Program,'' to the reporting requirements of TS Section 6.6.A.3, ''Steam Generator (SG) Tube Inspection Report,'' and to TS Sections 4.13 and 3.1.C, ''RCS [Reactor Coolant System] Operational Leakage.'' The proposed changes would establish alternate repair inspection and criteria for portions of the SG tubes within the tubesheet. The alternate inspection and repair criteria would be applicable to Unit 1 during Refueling Outage 23 (fall 2010) and the subsequent operating cycle and to Unit 2 during Refueling Outage 22 (fall 2009) and the subsequent operating cycle.

Date of issuance: November 5, 2009.

Effective date: Unit 1 is effective as of its date of issuance and shall be implemented by the end of the fall 2010 refueling outage. Unit 2 is effective as of its date of issuance and shall be implemented by the end of the fall 2009 refueling outage.

Amendment Nos.: 267 and 266.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the licenses and the technical specifications.

*Date of initial notice in **Federal Register:*** August 19, 2009 (74 FR 41939).

The supplements dated September 16, 2009 and September 30, 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. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 5, 2009.

No significant hazards consideration comments received: No.

Dated at Rockville, MD, this 19th day of November 2009.

For The Nuclear Regulatory Commission.

Joseph G. Giitter,

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

[FR Doc. E9–28630 Filed 11–30–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of November 30, December 7, 14, 21, 28, 2009, January 4, 2010.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

Week of November 30, 2009

Friday, December 4, 2009

9:30 a.m.—Meeting with the Advisory Committee on Reactor Safeguards (Public Meeting) (Contact: Antonio Dias, 301–415–6805).

This meeting will be Webcast live at the Web address—*http://www.nrc.gov.*

Week of December 7, 2009—Tentative

Tuesday, December 8, 2009

9:30 a.m.—Briefing on the Proposed Rule: Enhancements to Emergency, Preparedness Regulations (Public Meeting), (Contact: Lauren Quiñones, 301–415–2007).

This meeting will be Webcast live at the Web address—*http://www.nrc.gov.*

Week of December 14, 2009—Tentative

There are no meetings scheduled for the week of December 14, 2009.

Week of December 21, 2009—Tentative

There are no meetings scheduled for the week of December 21, 2009.

Week of December 28, 2009—Tentative

There are no meetings scheduled for the week of December 28, 2009.