7590-01-P

NUCLEAR REGULATORY COMMISSION

BIWEEKLY NOTICE

[NRC-2010-0156]

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 March 25, 2010 to April 7, 2010. The last biweekly notice was published on April 6, 2010 (75 FR 17439).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Non-timely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

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

accessing the documents located in ADAMS, should contact the NRC PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

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: November 30, 2009.

Description of amendment request: The amendments would revise Technical Specification (TS) 3.3.5, "Engineered Safety Features Actuation System Instrumentation," Table 3.3.5-1, to raise the refueling water tank (RWT) low level allowable values for the recirculation actuation signal (RAS); raise the minimum required RWT volume shown in TS Figure 3.5.5-1; and implement a time-critical operator action to close the RWT isolation valves, including consideration of a potentially more limiting single failure of a low-pressure safety injection pump to automatically stop, as designed, on an RAS.

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 RWT is a passive component of the Chemical and Volume Control System (CVCS) that supports ECCS [emergency core cooling system] and CSS [containment spray system] operation to mitigate the consequences of an accident. A[n] RAS is an active component of the Engineered Safety Features Actuation System (ESFAS) that actuates safety equipment to mitigate the consequences of a LOCA [loss-of-coolant accident]. Neither of these components initiates an accident previously evaluated. The RWT isolation valves are also components of

the CVCS; however, their closure was not previously credited for RWT isolation following a[n] RAS. The proposed amendment will credit closure of these valves following a[n] RAS to preclude the potential for air entrainment in the ECCS and CS [containment spray] pump suction piping for any LOCA scenario. The required isolation is being performed as a time critical operator action, which is consistent with ANSI/ANS-58.8-1984 [American National Standards Institute/American Nuclear Society Standard 58.8-1984], Time Response Design Criteria for Safety-Related Operator Actions, 1984 guidance. Although the change in the closure requirement and the operator action could introduce additional potential malfunctions, these malfunctions have been evaluated and found not to initiate or have a significant adverse affect on the mitigation or consequences of any accident previously evaluated.

The proposed changes do not alter or prevent the ability of structures, systems or components to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes will ensure continued performance of the ECCS and CS pumps following a LOCA by precluding the potential for air entrainment in the pump suction piping from the RWT after a[n] RAS.

The effect of the proposed changes to the RAS Allowable Values and RWT minimum required level on the RWT structural design, containment post-LOCA flood level, post-LOCA boron precipitation, and containment sump pH remain within the limits assumed in the design and accident analyses. The proposed license amendment does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite. The proposed license amendment is consistent with these analyses' assumptions and resultant consequences.

The proposed amendment also recognizes and evaluates a different single failure associated with the RWT drain down following a LOCA than previously evaluated. It was determined this failure was of low probability and did not adversely affect any previous bounding analysis or the capability of the associated systems to perform their design functions.

Therefore, the proposed license 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 license amendment does not involve or add any new or different components to the plant and does not change any accident initiators.

The proposed changes to the RAS Allowable Values and RWT minimum required level will not change the design function of the RWT to support ECCS and CSS operation following a LOCA. However, the closure of the RWT isolation valves following a LOCA was not previously credited. As a result, the credited RWT isolation valve design function has been changed, and closure of these valves is now credited to preclude the possibility of air entrainment in the ECCS and CS pump suction piping for any LOCA scenarios. The credited isolation is being performed as a time critical operator action, which is consistent with ANSI/ANS 58.8 guidance. Although changes to the valve closure requirement and the operator action introduce additional potential malfunctions, these malfunctions have been evaluated and found not to create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment recognizes and evaluates a different single failure associated with the RWT drain down following a LOCA than previously evaluated. It was determined that this failure was of low probability and did not adversely affect any previous bounding analysis or create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

The proposed license amendment does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined or implemented. The safety analysis acceptance criteria are not affected by this amendment. The proposed changes in the credited design function of the RWT isolation valves, along with the change in the RAS Allowable Value and RWT minimum required levels, continue to ensure sufficient RWT water volume to enable the ECCS and CSS to satisfy required design functions for all postulated LOCA break sizes. Therefore, these changes do not impact the results of safety analyses.

The proposed changes to the RAS Allowable Values and minimum required RWT level include appropriate instrument uncertainties and are based on conservative analyses for establishing the required RWT

volumes. The proposed amendment will not result in plant operation in a configuration outside of the design basis.

The proposed amendment recognizes and evaluates a different single failure associated with the RWT drain down following a LOCA than previously evaluated. It was determined this failure was of low probability and did not adversely affect any previous bounding analysis.

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 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.

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

Date of amendment request: January 29, 2010

Description of amendment request: The amendment would modify the existing Note within Technical Specification 3.4.10, “Pressurizer Safety Valves [PSVs],” which covers operation in the applicable portions of Mode 3.

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

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

No.

The proposed change, revising an existing NOTE within Technical Specification 3.4.10 to allow the PSVs lift settings to be outside LCO [Limiting Condition for Operation] values, as a result of temperature related drift, while the Unit is in applicable portions of Mode 3 for periods up to 36 hours, does not change the design function or operation of the PSVs and it does not change the way the PSVs are maintained, tested, or inspected. In addition the proposed change does not change any of the evaluated accidents in our Updated Final Safety Analysis Report, does not change PSV lift settings, or impact the ability of the PSVs to perform their safety function during evaluated accidents.

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

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

   No.

   The proposed change, revising an existing NOTE within Technical Specification 3.4.10 to allow the PSVs lift settings to be outside LCO values, as a result of temperature related drift, while the Unit is in applicable portions of Mode 3 for periods up to 36 hours, does not change the PSVs design function to maintain RCS [reactor coolant system] pressure below the RCS pressure Safety Limit of 2750 psia during design basis accidents nor does it affect the PSVs ability to perform this design function. The proposed change does not require any modification to the plant or change equipment operation or testing. It also does not create any credible new failure mechanisms, malfunctions, or accident initiators that would cause an accident not previously considered.

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

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

   No.

   The proposed change, revising an existing NOTE within Technical Specification 3.4.10 to allow the PSVs lift settings to be outside LCO values, as a result of temperature related drift, while the Unit is in applicable portions of Mode 3 for periods up to 36 hours, does not involve a significant reduction in the margin of safety in maintaining RCS pressure below Safety Limits of 2750 psia during design basis accidents. The analysis conducted in support of this proposed change evaluated the ability of the PSVs to maintain an adequate safety margin when required in applicable Mode 3 conditions despite the identified temperature related lift setting drift. The analysis identified that there were no credible design accident scenarios, when in the applicable Mode 3 conditions, that challenged the PSVs to respond in order to maintain an adequate safety margin to the reactor coolant Safety Limit of 2750 psia.

Therefore the proposed change does not involve a significant reduction in the margin of safety of maintaining RCS pressure the below RCS pressure Safety Limit.

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

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

NRC Branch Chief: Nancy L. Salgado.

Detroit Edison Company, Docket No. 50-341, Fermi 2, Monroe County, Michigan

Date of amendment request: January 4, 2010.

Description of amendment request: The proposed amendment would revise the Core Spray flow requirement in Technical Specifications Surveillance Requirements 3.5.1.8 and 3.5.2.6 from 6350 to 5725 gallons per minute consistent with the flow assumed in the Emergency Core Cooling System (ECCS) safety analyses.

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 minimum performance requirements of the low pressure Emergency Core Cooling System (ECCS) pumps, including the Core Spray pumps, are determined through application of the 10 CFR 50, Appendix K methodology to ensure the criteria of 10 CFR 50.46 are satisfied. The surveillance testing of the Core Spray pumps is performed periodically in accordance with the ASME Code, Section XI verifies that two Core Spray pumps in parallel operation within a single division develop sufficient discharge pressure at the Technical Specification required flow to overcome the elevation head pressure

between the pump suction and the vessel discharge, the piping friction losses, and TS SR specified Reactor Pressure Vessel pressure. The acceptance criteria necessary to satisfy the revised TS SRs would be established in the plant design basis in the form of the minimum required pump performance defined for a range of flow about the specified TS SR flow. Detroit Edison intends to continue TS SR and IST pump testing at the current IST pump baseline flow and establish compliance with the TS SR by comparing the measured performance against the design minimum pump curve. In this manner, the minimum actual delivered divisional Core Spray pump performance is assured to meet or exceed that required by the Appendix K safety analyses. These performance requirements are unchanged and are met by the proposed change.

The bases for the core spray flow requirements in the Technical Specifications Surveillance Requirements are unchanged. The requirements are selected based on the flow values assumed and used in the current ECCS safety analyses. The value proposed for core spray divisional (2 pump) flow is consistent with the inputs used for ECCS safety analyses performed for the current licensed power level.

Therefore, the proposed 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 revises the Technical Specification Surveillance Requirements for Core Spray flow to be consistent with the accident analysis. No physical changes are being made to the installed core spray system. The proposed surveillance requirements are consistent with those used in the accident analyses which analyze the effect of Core Spray system performance for the accident conditions for which the system is designed to respond. No new or different accident scenarios are created by this change.

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

3. The proposed change does not involve a significant reduction in the margin of safety.

   The Core Spray system has historically been capable of meeting the Core Spray Technical Specification Surveillance Requirements. However, correction of non-conservative errors in the system hydraulic calculation and the identification of a non-conservative bias in the test flow instrument calibration have eroded the test margin such that it is possible that the Technical Specification Surveillance Requirements may not be satisfied for some surveillances and at the same time maintain a relatively large margin compared to the minimum performance assumed in the ECCS safety analyses. These non-conservative errors or biases have always existed, but have not always been specifically accounted for in the surveillance testing acceptance criteria. Since there is no change in the Technical Specification bases associated with the requested change, there is no real change in the margin provided in the system design or analyses. The proposed change makes the margin between the current Core Spray Technical Specification

Surveillance Requirements and the performance assumed in the plant safety analyses available as a design and test margin. The minimum required performance necessary to satisfy the Core Spray Technical Specification Surveillance Requirements will be established in the plant design basis with the minimum required pump performance adjusted upward as necessary to account for instrument uncertainty and bias as well as differences between assumed accident and actual test operating conditions.

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: David G. Pettinari, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226-1279.

NRC Branch Chief: Robert J. Pascarelli.

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, as supplemented by letter dated March 18, 2010.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) requirements for testing of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Safety/Relief Valves (SRVs) by replacing the current requirement to manually actuate each SRV during plant startup with a requirement to verify that each valve is capable of being opened. The proposed amendment would change both TS Surveillance Requirements (SRs) 3.4.3.2 and 3.5.1.13 to verify that each required valve "is capable of being opened." The current

Frequency for both TS SRs is "24 months on a STAGGERED TEST BASIS for each valve solenoid"; this would be changed to state, "In accordance with the Inservice Testing Program."

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 change does not modify the method of demonstrating the Operability of the Safety/Relief Valves (SRVs) in both the safety and relief modes of operation. As currently stated in the Bases "...valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation." The proposed change does modify the method for demonstrating the proper mechanical functioning of the SRVs and that the valves and discharge lines are free of obstructions. The SRVs are required to function in the safety mode to prevent overpressurization of the reactor vessel and reactor coolant system pressure boundary during various analyzed transients, including Main Steam Isolation Valve closure. SRVs associated with the Automatic Depressurization System are also required to function in the relief mode to reduce reactor pressure to permit injection by low pressure Emergency Core Cooling System (ECCS) pumps during certain reactor coolant pipe break accidents. The current testing method demonstrates the proper mechanical functioning of the SRVs in both modes through manual actuation of the SRVs. The proposed new testing method demonstrates both Operability and proper mechanical functioning using a series of overlapping tests that demonstrate proper functioning of the SRV stages and supporting control components. This proposed testing method results in acceptable demonstration of the SRV functions in both the safety and relief modes, and therefore provides assurance that the probability of SRV failure will not increase. None of the accident safety analyses is affected by the requested Technical Specifications (TS) changes. Therefore, the consequences of accidents mitigated by the SRVs will not increase.

   Certain SRV malfunctions are included in the FSAR [final safety analysis report] safety analyses. Specifically, the plant safety analyses include the inadvertent opening of an SRV and a stuck open SRV. By not actuating the SRVs during plant operation for testing and thus reducing the

incidence of pilot stage leakage of the SRVs, the proposed testing eliminates a contributor to these events.

Based on these considerations, the proposed test method does 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 change modifies the method of testing of the SRVs, but does not alter the functions or functional capabilities of the SRVs. Testing under the proposed method is performed in offsite test facilities or in the plant during outage periods when the SRV functions are not required. Existing analyses address events involving an SRV inadvertently opening or failing to reclose. Analyses also address the likelihood and consequences of failure of one or more SRVs to open. The proposed change does not introduce any new failure mode, and therefore, does 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.

   Overpressure protection of the reactor coolant pressure boundary is based on the SRV setpoints and total relief capacity. Setpoint is verified at an offsite testing facility; this requirement is not altered by the proposed change. Relief capacity of each SRV is determined by valve geometry, which is also not altered by the test methods. The margin of safety in the Loss of Coolant Accident analysis due to operation of the Automatic Depressurization System is also based on total relief capacity of the associated SRVs. The proposed change in surveillance test methods demonstrates the operability of the SRVs, but does not alter the critical parameters that affect the margin of safety in analyses involving the SRV functions. 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: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: February 22, 2010.

Description of amendment request: The proposed amendment will allow implementation of leak-before-break (LBB) on the Waterford Steam Electric Station, Unit 3 (Waterford 3) pressurizer surge line. The licensee will be replacing the two Waterford 3 steam generators (SGs) during the forthcoming spring 2011 refueling outage. Based on design changes in the replacement SGs, piping systems will require rerouting in the SG cavity area. Due to the existing dynamic piping protection associated with the pressurizer surge line, rerouting of the replacement SG blowdown line cannot be effectively performed without the elimination of dynamic protection for the pressurizer surge line.

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 uses an approved leak-before-break (LBB) fracture mechanics methodology, in accordance with 10CFR50 [Title 10 of the *Code of Federal Regulations*, Part 50], Appendix A, General Design Criterion (GDC) 4 to demonstrate that the probability of fluid system rupture for these lines attached to the Reactor Coolant System (RCS) is

extremely low under conditions associated with the design basis for the piping. The proposed change does not adversely affect accident initiators or precursors nor significantly alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. Overall protection system performance will remain within the bounds of the previously performed accident analyses. The design of the protection systems will be unaffected. The Reactor Protection System (RPS) and Emergency Core Cooling System (ECCS) will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed amendment will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [Final Safety Analysis Report].

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 does not create the possibility of a new or different kind of accident, since it provides an NRC acceptable alternate means for demonstrating that the probability of a fluid system rupture is extremely small. There are no changes in the methods by which any safety-related plant system performs its safety function. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment. LBB methodology per GDC-4 still requires that ECCS, containment, and equipment qualification (EQ) requirements be maintained consistent with the original postulated accident assumptions. Only protection from dynamic effects is modified.

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 changes apply conservative approved analytical methods to demonstrate that the probability of a fluid system rupture is very low. This analysis retains substantial margins to assure that pipe rupture is

> extremely low and justifies differences in protection from dynamic effects with these extremely low probability ruptures. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. For overall ECCS, containment, and EQ requirements, there will be no changes to the assumed margins.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Joseph A. Aluise, Associate General Council - Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: February 22, 2010.

Description of amendment request: The proposed amendment would add valve SI-4052A (Reactor Coolant Loop (RCL) 2 Shutdown Cooling (SDC) suction inside containment bypass isolation) and valve SI-4052B (RCL 1 SDC suction inside containment bypass isolation) to Technical Specification (TS) Table 3.4-1, “Reactor Coolant System Pressure Isolation Valves.” The purpose of this line is to equalize the SDC system pressure down stream of valve SI-405A (RCL 2 SDC suction inside containment isolation) and valve SI-405B (RCL 1 SDC suction inside containment isolation) in order to minimize the pressure transient in the system when valves SI-405A(B) are opened.

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 bypass fill line will decrease the likelihood of a pressure transient in the Shutdown Cooling System suction piping which increases the reliability of the Shutdown Cooling System. Once this change is installed valves SI-405A(B) and SI-4052A(B) become parallel inside containment isolation valves in the shutdown cooling system suction lines. The configuration of SI-405A(B) and SI-4052A(B) includes interlocks such that these valves cannot be inadvertently opened with the RCS [reactor coolant system] above the design pressure of the shutdown cooling system. This change does not affect the capability of these valves to isolate the RCS from SDC. Therefore, there is no credible mechanism by which this change can introduce an inter-system LOCA [loss-of-coolant accident] (ISLOCA) different than previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. These features are, discussed in FSAR [Final Safety Analysis Report] section 7.6.1.1.2.

   Therefore, this 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.

   Once this change is installed valves SI-405A(B) and SI-4052A(B) become parallel inside containment isolation valves in the shutdown cooling system suction lines. SI-4052A(B) and its associated lines and valves are designed to the same requirements as SI-405A(B) and its associated lines. The previously evaluated SI-405A(B) failure modes bound those failure modes possible by SI-4052A(B). Thus, no failure of SI-4052A(B) exists that would be different or more severe than SI-405A(B),

   This 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 amendment adds SI-4052A(B) to Technical Specification Table 3.4-1. The change also adds an allowed leakage limit to SI-4052A(B) consistent with NUREG-1432 guidance.

Since the SI-4052A(B) leakage limit is commensurate with the valve size, this does not represent a significant reduction in a margin of safety.

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

Attorney for licensee: Joseph A. Aluise, Associate General Council – Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: February 22, 2010.

Description of amendment request: Entergy Operations, Inc. (the licensee), will be replacing the two Waterford Steam Electric Station, Unit 3 (Waterford 3) steam generators (SGs) during the 17th refueling outage which will commence in the spring of 2011. The existing Waterford 3 SG program under Technical Specification (TS) 6.5.9 contains an alternate repair criterion for SG tube inspections that is no longer applicable to the replacement SGs. The proposed amendment will modify TS 6.5.9, "Steam Generator (SG) Program," and TS 6.9.1.5, "Steam Generator Tube Inspection Report," to eliminate currently allowed SG tube alternate repair criteria and to modify the SG tube inservice inspection frequency.

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

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

   Response: No.

   The proposed change continues to implement the Waterford 3 Steam Generator Program performance criteria for tube structural integrity, accident induced leakage, and operational leakage for the replacement SGs. Meeting the performance criteria provides reasonable assurance that the replacement SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant system (RCS) pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident.

   The Steam Generator Tube Rupture (SGTR) is the primary accident analysis associated with SG tube integrity. The replacement SG tubing contains improved materials that will reduce the likelihood of tubing flaws. The proposed change to remove alternate repair criteria from the SG inspection program does not affect the design of the replacement SGs, their method of operation, operational leakage limits, or primary coolant chemistry controls. Therefore, the proposed change does not affect the probability of a SGTR accident. The SGs will be designed with substantial margin to burst. The SG tube inspection repair limit will also identify potential flaws before they become a safety concern. The extension of the SG tube inspection frequency after initial inspection is based on the low likelihood of having potential tube flaws and is considered to be an acceptable inspection period to preserve pressure boundary integrity. As a result, there will be no affect on the previous dose analysis reported in the FSAR [Final Safety Analysis Report] and the consequences of any accident are unchanged.

   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.

   Steam generator tube rupture events have already been postulated and analyzed in the Waterford 3 FSAR. The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary

coolant chemistry controls. Additionally, the proposed amendment does not impact any other plant systems or components. The TSs have established SG tube inspection requirements which assure that potential tubing flaws will be detected prior to affecting tube integrity and the RCS pressure boundary. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

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 structural integrity, accident induced leakage, and operational leakage performance criteria required by the Waterford 3 TSs provide substantial design margin for assuring SG tube integrity against the possibility of a SG tube pressure boundary failure. The proposed change removes an existing alternate repair criterion that is not applicable to the replacement SGs and establishes appropriate SG tube subsequent inspection periods consistent with the new SG tubing design. The replacement SGs will continue to meet their required performance criteria. The Waterford 3 SG tube inspection program will assure that this margin is maintained through the operational life of the plant.

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

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

Attorney for licensee: Joseph A. Aluise, Associate General Council – Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

Exelon Generation Company, LLC, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Units 1 and 2, Will County, Illinois

Date of amendment request: February 15, 2010.

Description of amendment request: This amendment request involves the adoption of Nuclear Regulatory Commission (NRC)-approved changes to the Standard Technical Specifications (STS) for Westinghouse plants (NUREG-1431), to allow relocation of specific TS surveillance frequencies to a licensee-controlled program. The proposed changes are described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b," as announced in the Notice of Availability published in the Federal Register on July 6, 2009 (74 FR 31996). Additionally, the proposed changes would add a new program, the Surveillance Frequency Control Program, to TS Section 5, Administrative Controls. The changes are applicable to licensees using the probabilistic risk guidelines contained in NRC-approved Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies."

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 adopted by the licensee is presented below:

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

   Response: No.

   The proposed changes relocate the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the Technical Specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance

criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

Response: No.

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

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

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

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated Final Safety Analysis Report and Bases to the Technical Specifications), because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant-licensing basis. To evaluate a change in the relocated surveillance frequency, EGC will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Revision 1 in accordance with the TS Surveillance Frequency Control Program. NEI 04-10, Revision 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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

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

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

NRC Branch Chief: Stephen J. Campbell.

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

Date of amendment request: February 15, 2010.

Description of amendment request: This amendment request involves the adoption of Nuclear Regulatory Commission (NRC)-approved changes to the Standard Technical Specifications (STS) for Westinghouse plants (NUREG-1431), to allow relocation of specific TS surveillance frequencies to a licensee-controlled program. The proposed changes are described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b," as announced in the Notice of Availability published in the Federal Register on July 6, 2009 (74 FR 31996). Additionally, the proposed changes would add a new program, the Surveillance Frequency Control Program, to TS Section 5, Administrative Controls. The changes are applicable to licensees using the probabilistic risk guidelines contained in NRC-approved Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies."

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 adopted by the licensee is presented below:

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

   Response: No.

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

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

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

   Response: No.

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

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

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

   Response: No.

   The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards

> (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated Final Safety Analysis Report and Bases to the Technical Specifications), because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant-licensing basis. To evaluate a change in the relocated surveillance frequency, EGC will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Revision 1 in accordance with the TS Surveillance Frequency Control Program. NEI 04-10, Revision 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.
>
> Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Stephen J. Campbell.

Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station (DNPS), Units 2 and 3, Grundy County, Illinois

Date of amendment request: February 4, 2010.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.3.61, “Primary Containment Isolation Instrumentation,” Table 3.3.6.1-1, “Primary Containment Isolation Instrumentation,” Function 6.a, “Shutdown Cooling System Isolation, Recirculation Line Water Temperature - High,” to enable implementation of a

modification that replaces the temperature-based isolation instrumentation with reactor pressure-based isolation instrumentation. The proposed modification will address instrumentation reliability problems that have led to interruptions of Shutdown Cooling (SDC) system operation, leading to unplanned heat-up of reactor coolant while the reactor was in operational Modes 3 and 4.

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 license amendment implements a revised process parameter and the associated Allowable Value (AV) for the DNPS Units 2 and 3 SDC system isolation function 6.a in TS Table 3.3.6.1-1.

The proposed changes to the isolation function do not affect the probability of any event initiators at the facilities. This isolation function is provided for equipment protection to prevent exceeding the system design temperature. The isolation function is not credited or assumed in the accident or transient analysis in the Updated Final Safety Analysis Report (UFSAR).

The proposed changes will not degrade the performance of, or increase the number of challenges imposed on, safety-related equipment that is assumed to function during an accident situation. The SDC system and the isolation function that is being revised are not safety related and are not credited to function during an accident situation. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the UFSAR.

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 license amendment implements a revised process parameter and AV for the DNPS Units 2 and 3 SDC system isolation function 6.a in TS Table 3.3.6.1-1.

The proposed change enables implementation of a modification that will enhance the reliability of instrumentation used to protect the functionality and integrity of the non safety-related SDC system. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. 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 license amendment revises a process parameter and AV for the DNPS Units 2 and 3 SDC system isolation function 6.a in TS Table 3.3.6.1-1.

The margin of safety is established through the design of the plant structures, systems, and components (SSCs), the parameters within which the plant is operated, and the setpoints for the actuation of equipment relied upon to respond to an accident.

The proposed change to the SDC system isolation instrumentation function for the SDC system does not change the SSCs, operational parameters, or actuation setpoints for equipment that is relied upon to respond to an accident. Both the SDC system and the isolation function that is being revised are non-safety related and are not credited to function during an accident situation.

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station (DNPS), Units 2 and 3, Grundy County, Illinois

Date of amendment request: February 16, 2010.

Description of amendment request: The proposed amendments would modify the DNPS Units 2 and 3, Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)-425, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b," Revision 3. Additionally, the change would add a new program, the "Surveillance Frequency Control Program [SFCP]," to TS Section 5, "Administrative Controls."

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 licensee reviewed the proposed No Significant Hazards Consideration (NSHC) determination published in the Federal Register dated July 6, 2009 (74 FR 31996). The licensee has concluded that the proposed NSHC presented in the Federal Register notice is applicable to DNPS, Units 2 and 3. The proposed NSHC is presented below:

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

   Response: No.

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

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

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

Response: No.

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

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

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

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to the TS), because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, EGC will utilize the guidance contained in NRC-approved NEI 04-10, in accordance with the TS SFCP. NEI 04-10, Revision 1 methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: February 15, 2010.

Description of amendment request: The proposed amendments would modify the LaSalle County Station (LSCS) Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10.

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:

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

   Response: No.

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

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

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

   Response: No.

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

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

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

   Response: No.

   The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the Updated Final Safety Analysis Report and Bases to the Technical Specifications), because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, EGC will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Revision 1 in accordance with the TS Surveillance Frequency Control Program. NEI 04-10, Revision 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: February 22, 2010.

Description of amendment request: The proposed amendments would revise Technical Specification 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time associated with Condition B from 8 hours to 72 hours.

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

   The proposed change is based on a risk-informed evaluation performed in accordance with Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications."

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

   The proposed amendment does not increase postulated frequencies or the analyzed consequences of an Anticipated Transient Without Scram (ATWS). Requirements associated with 10 CFR 50.62 will continue to be met. In addition, the proposed amendment does not increase postulated frequencies or the analyzed consequences of a large-break loss-of-coolant accident for which the SLC system will be used for pH control (i.e., upon NRC approval of an August 26, 2008 proposed LSCS license amendment regarding the adoption of an alternate source term methodology). The extended CT provides additional time to implement actions in response to a dual-train SLC system inoperability, while also minimizing the risk associated with continued operation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

The proposed amendment revises TS 3.1.7 to extend the CT associated with Condition B from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. This proposed TS amendment does not change the design function of the SLC system and does not affect the system's ability to perform its design function. The SLC system provides a method to bring the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. Required actions and surveillance requirements are sufficient to ensure that the SLC system functions are maintained. No new accident initiators are introduced by this amendment. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The proposed amendment revises TS 3.1.7 to extend the CT associated with Condition B from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the setpoints for the actuation of equipment relied upon to respond to an event.

   Safety margins applicable to the SLC system include pump capacity, boron concentration, boron enrichment, and system response timing. The proposed amendment does not modify these safety margins or the point at which SLC is manually initiated, nor does it affect the system's ability to perform its design function. In addition, the proposed change complies with the intent of the defense-in-depth philosophy and the principle that sufficient safety margins are maintained, consistent with RG 1.177 requirements (i.e., Section C, "Regulatory Position," paragraph 2.2, "Traditional Engineering Considerations").

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

Exelon Generation Company, LLC, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, Rock Island County, Illinois

Date of amendment request: February 16, 2010.

Description of amendment request: The proposed amendments would modify the QCNPS Units 1 and 2, Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)-425, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b," Revision 3. Additionally, the change would add a new program, the "Surveillance Frequency Control Program [SFCP]," to TS Section 5, "Administrative Controls."

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 licensee reviewed the proposed No Significant Hazards Consideration (NSHC) determination published in the Federal Register dated July 6, 2009 (74 FR 31996). The licensee has concluded that the proposed NSHC presented in the Federal Register notice is applicable to QCNPS, Units 1 and 2. The proposed NSHC is presented below:

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

   Response: No.

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

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

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

   Response: No.

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

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

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

   Response: No.

   The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to the TS), because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, EGC will utilize the guidance contained in NRC-approved NEI 04-10, in accordance with the TS SFCP. NEI 04-10, Revision 1 methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Stephen J. Campbell.

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: December 14, 2009.

Description of amendment request: The proposed amendment would remove the structural integrity requirements contained in Technical Specifications (TSs) 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) and their associated Bases; incorporate changes to accident monitoring instrumentation for consistency with NUREG-1432 actions and allowed outage times for conditions that drive a unit to hot shutdown; and administrative corrections based on obvious typos, previous amendments, or obsolete requirements.

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 to remove structural integrity controls from the TSs does not impact any mitigation equipment or the ability of the RCS [reactor coolant system] pressure boundary to fulfill any required safety function. The proposed change will continue to ensure the requirements of 10 CFR 50.55a are maintained as specified in TS 4.0.5 and the new administrative TS program for RCP [reactor coolant pump] flywheel inspections. The changes to the accident instrumentation actions and allowed outage time have no appreciable effect on accident initiation or mitigation. Since no other accident mitigation or initiators are impacted by this change, no design basis accidents are affected.

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

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

   The proposed change will not alter the plant configuration or change the manner in which the plant is operated. Structural integrity will continue to be maintained as required by 10 CFR 50.55a and specified in TS 4.0.5 and the new administrative TS program for RCP flywheel inspections. Accident monitoring instrumentation does not contribute to failure modes. No new failure modes are being introduced by the proposed change.

   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 the margin of safety?

   Removing TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) from the TSs does not reduce the controls that are required to maintain the structural integrity of ASME Code Class 1, 2, or 3 components. There is no increase with any accident mitigation risk associated with the accident monitoring instrumentation TS changes as the proposed allowed outage times and the intervening step through HOT STANDBY are consistent with the equivalent to NUREG-1432 completion times and actions for post accident instrumentation and are equal to or more conservative than the current TS requirements. No other safety margins are impacted due to the proposed change.

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

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

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

NRC Acting Branch Chief: Douglas A. Broaddus.

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

Date of amendment request: February 25, 2010.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement (SR) 3.8.1.9, Diesel Generator (DG) Load Test, to correct a non-conservative power factor (PF) value and to add a new note consistent with TS Task Force (TSTF) traveler TSTF-276-A, Revision 2, "Revise DG Full Load Rejection Test."

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

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

   Response: No.

   Performing a surveillance that tests the DG is not a precursor of any accident previously evaluated. Revising the PF limit to be more conservative, and relaxing the requirement to maintain PF when paralleled to offsite power does not significantly affect the method of performing the surveillances such that the probability of an accident would be affected. These changes only affect surveillances of mitigative equipment and, therefore, do not have an impact on the probability of an accident previously evaluated.

   Revising the surveillances by specifying a more conservative PF value ensures the DG's will provide the power assumed in calculations of design basis accident mitigation. Relaxing the requirement to maintain PF when paralleled to offsite power does not affect performance of the DG under accident conditions. The performance of the surveillances ensures that mitigative equipment is capable of performing its intended function, and therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The 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 a safety-related system or component and do not challenge the performance or integrity of safety related systems. As such, it does not introduce a mechanism for initiating a new or different accident than those described in the USAR [updated safety analysis report].

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 changes will continue to ensure the DGs are able to perform their design function as assumed in calculations that evaluate their function during design basis accidents. Decreasing the PF limit for testing will not affect the design or functioning of the DGs. The increased reactive loading required to maintain the PF below the limit is small and well within DG capability. Based on this, the ability of CNS [Cooper Nuclear Station] to mitigate the design basis accidents that rely on operation of the DG's is not adversely impacted. Revising the PF increases the margin of safety by specifying a more conservative value for the PF limit. Therefore, NPPD [Nebraska Public Power District] concludes these proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, 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. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602-0499.

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.

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

Date of application for amendment: July 13, 2007, as supplemented by letters dated. July 13, 2007, September 30, 2008, March 5, 2009, March 23, 2009, March 1, 2010, and March 5, 2010.

Brief description of amendment: The license amendment revises the Millstone Power Station, Unit No. 3 (MPS3) spent fuel pool (SFP) storage requirements. The July 13, 2007, license amendment request proposed a stretch power uprate (SPU) of MPS3. Included in a supplement dated July 13, 2007, was a request to amend the MPS3 SFP storage requirements. The July 13, 2007, request was noticed in the *Federal Register* on January 15, 2008 (73 FR 2549). By letter dated March 5, 2008, Dominion Nuclear Connecticut, Inc. (DNC) separated the MPS3 SFP storage requirements request from the MPS3 SPU request. The request to revise the MPS3 SFP storage requirements was re-noticed on September 8, 2009 (74 FR 46241) using the original significant hazards consideration, specific to the request to revise the SFP storage.

Date of issuance: March 26, 2010.

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

Amendment No.: 248.

Renewed Facility Operating License No. NPF-49: Amendment revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: January 15, 2008 (73 FR 2549) and September 8, 2009 (74 FR 46241). The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination as published in the *Federal Register* (73 FR 2549). The SFP LAR no significant hazards consideration determination was noticed a second time, separate from the MPS3 SPU (74 FR 46241).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 2010.

No significant hazards consideration comments received: No.

Entergy Gulf States Louisiana, LLC, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1 (RBS), West Feliciana Parish, Louisiana

Date of amendment request: June 29, 2010.

Brief description of amendment: The amendment revised the RBS Technical Specification (TS) 5.5.6, “Inservice Testing Program.” TS 5.5.6 contains references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI as the source for the inservice testing (IST) of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes delete the references to Section XI of the ASME Code and incorporate references to

the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). In addition, the amendment changes will limit applying Surveillance Requirement (SR) 3.0.2 to surveillances with a frequency of 2 years or less. These changes are consistent with the changes identified in the Improved Standard Technical Specifications (ISTS) in Technical Specification Task Force Traveler (TSTF) Change Travelers TSTF-479, “Changes to Reflect Revision of 10 CFR 50.55a," and TSTF-497, "Limit Inservice Testing Program 3.0.2 Application to Frequencies of 2 Years or Less."

Date of issuance: March 31, 2010.

Effective date: As of the date of issuance and shall be implemented 90 days from the date of issuance.

Amendment No.: 167.

Facility Operating License No. NPF-47: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: August 25, 2009 (74 FR 42928).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 27, 2009.

Brief description of amendment: The amendment revised Technical Specification (TS) Section 2.1.1, "Reactor Core SLs [Safety Limits]," Subsection 2.1.1.2, to change the two recirculation loop safety limit for minimum critical power ratio (SLMCPR) from 1.08 to 1.09 and the single recirculation loop SLMCPR from 1.10 to 1.12. The changes to the TSs are necessary as a result of the GGNS Cycle 18 cycle-specific SLMCPR calculations.

Date of issuance: March 25, 2010.

Effective date: As of the date of issuance and shall be implemented after the current cycle (Cycle 17) is completed and prior to the operation of Cycle 18.

Amendment No: 184.

Facility Operating License No. NPF-29: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: January 5, 2010 (75 FR 461).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 25, 2010.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Units 1 and 2 (Braidwood), Will County, Illinois

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

Date of application for amendment: December 4, 2008, as supplemented by letters dated February 17, 2009; July 27, 2009; December 4, 2009; and January 29, 2010.

Brief description of amendment: The amendments revise Technical Specifications (TSs) 1.1, "Definitions," and 3.4.16, "RCS [Reactor Coolant System] Specific Activity," and Surveillance Requirements 3.4.16.1, 3.4.16.2, and 3.4.16.3. The revisions replace the current TS 3.4.16 limit on RCS gross specific activity with a new limit on RCS noble gas-specific activity. The revisions adopt TS Task Force (TSTF) Change Traveler, TSTF-490, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec [sic],"
Revision 0.

Date of issuance: March 23, 2010.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment Nos.: Braidwood Unit 1 - 162; Braidwood Unit 2 -162; Byron Unit No. 1 -167; and Byron Unit No. 2 -167.

Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66: The amendments revise the TSs and Licenses.

Date of initial notice in FEDERAL REGISTER: January 27, 2009 (74 FR 4771).

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 23, 2010.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station (DNPS), Units 2 and 3, Grundy County, Illinois

Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: April 7, 2009, as supplemented by letter dated October 5, 2009.

Brief description of amendments: The amendments delete a footnote from DNPS Technical Specification (TS) 3.4.5, “RCS Leakage Detection Instrumentation,” that was incorporated as part of a limited duration emergency license amendment in August 2008, and is no longer applicable. The amendments also correct errors in the titles of analytical methods in DNPS and QCNPS TS 5.6.5, "Core Operating Limits Report (COLR)," paragraph b. The proposed changes delete historical analytical methods from DNPS and QCNPS TS 5.6.5.b that are no longer applicable, and renumber the remaining analytical methods.

Date of issuance: April 1, 2010.

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

Amendment Nos.: 234/227, 246/241.

Renewed Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications and License.

Date of initial notice in FEDERAL REGISTER: June 30, 2009 (74 FR 31322).

The October 5, 2009, supplement, contained clarifying information and did not change the NRC staff’s initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 1, 2010.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: September 28. 2009, as supplemented by letter dated January 20, 2010.

Brief description of amendment request: The proposed amendment would support application of optimized weld overlays or full structural weld overlays. Applying these weld overlays on the reactor coolant pump suction and discharge nozzle dissimilar metal welds requires an update to the DBNPS leak-before-break (LBB) evaluation.

Date of issuance: March 24, 2010.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 281.

Facility Operating License No. NPF-3: The amendment revised the current licensing basis.

Date of initial notice in FEDERAL REGISTER: February 22, 2010 (75 FR 7628).

The January 20, 2010 supplement, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 24, 2010.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50-302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: November 6, 2008; superseded by letters dated August 4 and December 4, 2009.

Brief description of amendment: The amendment modifies the Crystal River Unit 3 (CR-3) technical specifications (TS) surveillance requirements (SRs) related to allowable voltage and frequency limits for the emergency diesel generator (EDG) testing. Specifically, the amendment revises the CR-3 TS SRs 3.8.1.2, 3.8.1.6, 3.8.1.10.c.3 and 3.8.1.10.c.4 to restrict the voltage and frequency limits for both slow and fast EDG starts.

Date of issuance: December 10, 2009.

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

Amendment No. 236.

Facility Operating License No. DPR-72: Amendment revises the facility operating license and the technical specifications.

Date of initial notice in *Federal Register*: September 8, 2009 (74 FR 46242). The supplement dated December 4, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated December 10, 2009.

No significant hazards consideration comments received: No.

NextEra Energy Duane Arnold, LLC, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: March 4, 2009.

Brief description of amendment: The amendment changed the Duane Arnold Energy Center Technical Specification (TS) Section 5.5.12 (Primary Containment Leakage Rate Testing Program) to exclude the Main Steam pathway leakage contribution from the overall integrated leakage rate Type A test measurement and from the sum of the leakage rates from Type B and Type C tests and changed TS Section 3.6.1.3 (Primary Containment Isolation Valves) to remove the repair criterion for main steam isolation valves that fail their as-found leakage rate acceptance criterion found in current Surveillance Requirement 3.6.1.3.9.

Date of issuance: March 31, 2010.

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

Amendment No.: 276.

Facility Operating License No. DPR-49: The amendment revised the Technical Specifications.

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 March 31, 2010.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50-410, Nine Mile Point Nuclear Station (NMPNS), Unit No. 2 (NMP2), Oswego County, New York

Date of application for amendment: June 29, 2009, as supplemented on August 13, 2009, and February 3, 2010.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.12, “10 CFR 50 Appendix J Testing Program Plan,” by replacing the reference to Regulatory Guide 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A,

as the implementation document used by NMPNS to develop the NMP2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. In addition, the amendment allows NMPNS to extend the current interval for the NMP2 primary containment integrated leak rate test (ILRT) from 10 years to 15 years, and allows successive ILRTs to be performed at 15-year intervals.

Date of issuance: March 30, 2010.

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

Amendment No.: 134.

Renewed Facility Operating License No. NPF-069: The amendment revises the License and TSs.

Date of initial notice in FEDERAL REGISTER: October 20, 2009 (74 FR 53779).

The supplemental letters dated August 13, 2009, and February 3, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 30, 2010.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of application for amendment: October 8, 2009, as supplemented by letter dated February 25, 2010.

Brief description of amendments: The amendment approves a one-time change to Technical Specification (TS) 6.8.4.i, “Steam Generator (SG) Program,” regarding the SG tube inspection and repair required for the portion of the SG tubes passing through the tubesheet region. Specifically, for Salem Unit No. 1 refueling outage 20 (planned for spring 2010) and subsequent operating cycles until the next scheduled SG tube inspection, the amendment limits the required inspection (and repair if degradation is found) to the portions of the SG tubes passing through the upper 13.1 inches of the approximate 21-inch tubesheet region. In addition, the amendment revises TS 6.9.1.10, “Steam Generator Tube Inspection Report,” to provide reporting requirements specific to the one-time change.

Date of issuance: March 29, 2010.

Effective date: As of the date of issuance, to be implemented prior to completion of refueling outage 20 (currently scheduled for spring 2010).

Amendment No.: 294.

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

Date of initial notice in FEDERAL REGISTER: January 5, 2010 (75 FR 464).

The letter dated February 25, 2010, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original *Federal Register* notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 29, 2010.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment requests: February 3, 2009, and March 3, 2009; both applications were supplemented by letters dated November 20, 2009, and January 20, 2010.

Brief description of amendments: The amendments approved a revision to the South Texas Project (STP), Units 1 and 2 Fire Protection Program for Fire Areas 27 and 31. In the event of a fire in the Fire Areas 27 and 31, the amendments allow the licensee to perform operator manual actions to achieve and maintain safe shutdown in lieu of meeting the circuit separation and protection requirements of Title 10 of the *Code of Federal Regulations,* Part 50, Appendix R, Section III.G.2. The amendments revised the License Condition 2.E, "Fire Protection," in the facility operating licenses, to reflect the changes. The approved changes to the Fire Protection Program will be documented in the licensee's "Fire Hazards Analysis Report."

Date of issuance: March 31, 2010.

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

Amendment Nos.: Unit 1 - 193; Unit 2 - 181.

Facility Operating License Nos. NPF-76 and NPF-80: The amendments revised the Facility Operating Licenses.

Date of initial notices in *Federal Register*: August 25, 2009 (74 FR 42929, 42930). The supplemental letters dated November 20, 2009, and January 20, 2010, provided additional information that clarified the applications, did not expand the scope of the applications as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 2010.

No significant hazards consideration comments received: No.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION AND OPPORTUNITY FOR A HEARING (EXIGENT PUBLIC ANNOUNCEMENT OR EMERGENCY CIRCUMSTANCES)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a *Federal Register* notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to

the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 51.12(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 application for amendment, (2) the amendment to Facility Operating License, 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's (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.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/doc-collections/cfr/. If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397-4209, (301) 415-4737, or by e-mail to pdr.resource@nrc.gov. 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 petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the 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.[1] 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 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.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. Technical - - primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.

2. Environmental - - primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.

3. Miscellaneous - - does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/requestors shall jointly designate a representative who shall have the authority to act for the petitioners/requestors with respect to that contention. If a requestor/petitioner seeks to adopt the contention of another sponsoring requestor/petitioner, the requestor/petitioner who seeks to adopt the contention must either agree that the sponsoring requestor/petitioner shall act as the representative with respect to that contention, or jointly designate with the sponsoring requestor/petitioner a representative who shall have the authority to act for the petitioners/requestors with respect to that contention.

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[1]To the extent that the applications contain attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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.

Carolina Power and Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina.

Date of amendment request: March 22, 2010, as supplemented on March 23, 2010.

Description of amendment request: The previous Technical Specification (TS) 3.4.17, "Chemical and Volume Control System (CVCS)," Action B, allowed the licensee 24 hours to restore an inoperable makeup water pathway from the Refueling Water Storage Tank before taking further actions. This amendment increased the completion time of TS 3.4.17, Action B, from 24 hours to 72 hours for fuel cycle 26.

Date of issuance: March 25, 2010.

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

Amendment No.: 223.

Facility Operating License No. (DPR-23): Amendment revises the technical specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC):

No. The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated March 25, 2010.

Attorney for licensee: David T. Conley, Associate General Counsel II - Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Douglas A. Broaddus.

Dated at Rockville, Maryland, this 12th day of April, 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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