need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g., braille, large print), please notify the NRC's Disability Program Coordinator, Rohn Brown, at 301–492–2279, TDD: 301–415–2100, or by e-mail at *REB3@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: November 29, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–5936 Filed 11–30–07; 10:19 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 8, 2007 to November 21, 2007. The last biweekly notice was published on November 20, 2007 (72 FR 65360).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license, and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene.

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

Written comments may be submitted by mail to the Chief, Rulemaking, Directives and Editing Branch, 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 delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, person(s) may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license, and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request via electronic submission through the NRC E-Filing system for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for hearing or a petition for leave to intervene must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated on August 28, 2007, (72 FR 49139). The E-Filing process requires participants to submit and serve documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *HEARINGDOCKET@NRC.GOV,* or by calling (301) 415–1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer(tm) to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at *http://www.nrc.gov/site-help/e-submittals/install-viewer.html.* Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.*

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The EIE system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the ''Contact Us'' link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html* or by calling the NRC technical help line, which is available between 8:30 a.m. and 4:15 p.m., Eastern Time, Monday through Friday. The help line number is (800) 397–4209 or locally, (301) 415–4737.

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

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii). To be timely, filings must be submitted no later than

11:59 p.m. Eastern Time on the due date.

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

For further details with respect to this amendment action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendments request: January 22, 2007, as supplemented by letters dated June 21, July 18, July 31, and October 15, 2007.

Description of amendments request: The amendment would revise the Technical Specifications to support the transition to AREVA NP fuel and core design methodologies.

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

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

Response: No.

The proposed amendments revise the list of NRC-approved analytical methods used to establish core operating limits. Core operating limits are established to ensure that fuel design limits are not exceeded during operating transients or accidents. The analytical methods used to determine core operating limits are those methods that have previously been found acceptable by the NRC and are required to be listed in the Technical Specification section governing the Core Operating Limits Report. The application of these NRC-approved analytical methods will continue to ensure that acceptable operating limits are established and applied to operation of the reactor core.

The proposed amendments will add a new Technical Specification 3.2.3, "Linear Heat Generation Rate (LHGR)," for fuel bundles, add a new definition to Technical Specification 1.1 for LHGR, and revise Technical Specifications 3.4.1 and 3.7.6 to incorporate restrictions on LHGR when in single recirculation loop operation or with an inoperable Turbine Bypass System. These LHGR limits will be established using NRC-approved analytical methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable.

Based on the above, the proposed amendments do not involve an 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.

As previously stated, the proposed amendments support transition from Global Nuclear Fuels Americas (GNF–A) fuel and core design and analysis services to AREVA NP fuel and core design and analysis services. The AREVA NP fuel assemblies which will be used in the BSEP Unit 1 and 2 cores will be similar in design to the GNF–A fuel that will be co-resident in the cores. The BSEP, Unit 1 and 2 cores in which this fuel will operate will be designed to meet all applicable design and licensing criteria. Adherence to these design and licensing criteria will not introduce any new modes of operation or introduce any new accident precursors, and thus will preclude the introduction 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 amendments will continue to require that core operating limits be determined using NRC-approved analytical methods. Acceptable fuel performance is obtained by ensuring that the peak cladding temperature (PCT) during a postulated design basis loss-of-coolant accident (LOCA) is maintained less than the limits specified in 10 CFR 50.46, and that the core remains in a coolable geometry following a postulated design basis LOCA. The proposed amendments ensure that adequate margin will continue to be maintained to the 2200 degree PCT limit of 10 CFR 50.46, and the use of NRC-approved analytical methods will continue to ensure acceptable fuel performance during normal operations, as well as during transient and accident conditions. Therefore, the proposed amendments 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: 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: Thomas H. Boyce.

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

Date of amendments request: August 6, 2007.

Description of amendments request: The amendment would revise the Technical Specifications (TSs) to implement Technical Specification Task Force (TSTF) Change TSTF–343, Revision 1, which allows the performance of visual examinations of the primary containment to be performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL. The amendment would also make an administrative change to the TSs by eliminating a one-time requirement to perform containment leak rate testing that has already been completed.

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 affects the frequency of visual examinations that will be performed for the concrete surfaces of the containment for the purpose of the Primary Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the metallic and concrete surfaces of the containment and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed in accordance with NRC-approved ASME Section XI Code requirements, except where relief has been granted by the NRC, to meet the intent of visual examinations specified by Regulatory Guide 1.163, without requiring additional visual examinations in accordance with the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more vigorous requirements of the Code-required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained.

The proposed change also includes the removal of an item in TS 5.5.12 which was incorporated to establish deadlines for performing the performance-based Type A leakage tests in conjunction with changing, on a one-time basis, the Type A test frequency. The specified Unit 1 and Unit 2 Type A test have been completed. As such, removal of this item is an administrative change.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. The proposed change does not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, based on the above, the proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change revises the Primary Containment Leakage Rate Testing Program in TS 5.5.12 for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class MC and CC. The proposed change affects the frequency of visual examinations that will be performed for the metallic and concrete surfaces of containment and allows those examinations to be performed during power operation as opposed to during a refueling outage.

The proposed change does not involve a modification to the physical configuration of the plants (i.e., no new equipment will be installed), and does not revise the methods governing normal plant operation. Also, the proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism.

The proposed change also includes the removal of an item in TS 5.5.12 which was incorporated to establish deadlines for performing the performance based Type A leakage tests in conjunction with changing, on a one-time basis, the Type A test frequency. The specified Unit 1 and Unit 2 Type A test have been completed. As such, removal of this item is an administrative change.

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

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

Response: No.

The proposed change revises the Primary Containment Leakage Rate Testing Program in TS 5.5.12 for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class MC and CC. The proposed change allows some of those examinations to be performed during power operation as opposed to during a refueling outage. As previously stated, the proposed change does not involve a modification to the physical configuration of the plants and does not revise the methods governing normal plant operation. As such, the safety function of the containment as a fission product barrier, will be maintained and is not adversely impacted by the proposed change.

The proposed change also includes the removal of an item in TS 5.5.12 which was incorporated to establish deadlines for performing the performance-based Type A leakage tests in conjunction with changing, on a one-time basis, the Type A test frequency. The specified Unit 1 and Unit 2 Type A test have been completed. As such, removal of this item is an administrative change.

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 T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, NC 27602.

NRC Branch Chief: Thomas H. Boyce.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336 Millstone Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: February 20, 2007.

Description of amendment request: The proposed amendment would revise the Millstone Power Station, Unit No. 2 (MPS2) Technical Specifications (TS) to eliminate Surveillance Requirement (SR) 4.5.2.e which requires flow rate verification for each charging pump. Charging pump flow is no longer relied upon for design basis mitigation at MPS2 and the charging pumps have been classified as non-risk significant in the MPS2 Probabilistic Risk Assessment model. Therefore, the proposed amendment is requesting to remove the charging pump flow verification requirements currently located in the TS SR 4.5.2.e.

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 FSAR [Final Safety Analysis Report] Chapter 14 accident analyses for MPS2 take no credit for the flow delivered by the charging pumps. Additionally, the proposed change does not modify any plant equipment or method of operation for any system, structure or component required for safe operation of the facility or mitigation of accidents assumed in the facility safety analyses. As such, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not modify any plant equipment or method of operation for any system, structure or component required for safe operation of the facility or mitigation of accidents assumed in the facility safety analyses. As such, no new failure modes are introduced by the proposed change. Consequently, the proposed amendment does not introduce any accident initiators or malfunctions that would cause a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The FSAR Chapter 14 accident analyses for MPS2 take no credit for the charging pumps. The TS change does not involve a significant reduction in a margin of safety because the proposed change does not affect equipment design or operation, and there are no changes being made to the technical specification required safety limits or safety system settings. The proposed change does not affect any of the assumptions used in the accident analysis, nor does it affect any method of operation for equipment important to plant safety. Therefore, the margin of safety is not impacted by the proposed amendment.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Branch Chief: Harold K. Chernoff.

Dominion Nuclear Connecticut Inc., et al., Docket Nos. 50–336 and 50–423, Millstone Power Station, Unit Nos. 2 and 3, New London County, Connecticut.

Date of amendment request: July 2, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) 4.0.5 to reference the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) instead of Section XI of the ASME *Boiler and Pressure Vessel Code.* Specifically, the proposed amendment would modify the inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves to reflect the requirements in the ASME OM Code. In addition, the redundant requirement in TS 4.0.5 to maintain an ISI program is being proposed for removal, based on duplicate regulatory requirements set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a.

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

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

Response: No.

The proposed change does not modify any plant equipment and does not impact any failure modes that could lead to an accident. Additionally, the proposed change has no effect on the consequence of any analyzed accident since the change does not affect the function of any equipment credited for accident mitigation. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. Removing from TS the duplicate requirement in the regulations to maintain an ISI program in accordance with ASME codes and standards does not impact any accident initiators or analyzed events or mitigation of events. No reduction in previous commitments to 10 CFR 50.55a(g) are being proposed by this change.

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

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

Response: No.

The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or adversely affect methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. The proposed change does not alter existing test criteria or frequencies. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. Removal of the duplicate TS requirement to maintain an ISI program will not alter the commitment to the current ISI program requirements in 10 CFR 50.55a or any other TS requirements related to inservice inspection.

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

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

Response: No.

The proposed change revises TS 4.0.5 regarding inservice testing of ASME Code Class 1, 2, and 3 pumps and valves, for consistency with the requirements of 10 CFR 50.55a(f)(4). The proposed change incorporates an administrative clarification to the frequencies for IST and incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. No setpoints or safety limit settings are being revised. The safety function of the affected pumps and valves will continue to be confirmed through inspection and testing. Removal of the ISI program requirement from TS 4.0.5 does not remove the requirement from regulations, and therefore, will not diminish the current station approved programs and procedures that implement the regulatory criteria of 10 CFR 50.55a(g) to maintain an acceptable ISI program in accordance with the ASME Code.

Based on the discussion above, 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: Lillian M. Cuoco, Esquire, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Building 475, 5th Floor, Rope Ferry Road, Waterford, CT 06141–5127.

NRC Branch Chief: Harold K. Chernoff.

Duke Power Company LLC, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

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

Duke Power Company LLC, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Oconee Nuclear Station Independent Spent Fuel Storage Installation NRC License No. SNM–2503, Docket No. 72–4, Oconee County, South Carolina

Date of amendment request: March 14, 2007.

Description of amendment request: The amendments would revise the licenses to reflect the change in the name of the licensee from Duke Power Company LLC to Duke Energy Carolinas, LLC. The proposed amendments are a name change only. There is no change in the state of incorporation, registered agent, registered office, rights or liabilities of the company. Nor is there a change in the function of the licensee or the way in which it does business.

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 amendments are for a name change only. The amendments do not involve any change in the technical qualifications of the licensee or the design, configuration, or operation of the nuclear units. All Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications remain unchanged. Also, the Physical Security Plans and related plans, the Operator Training and Requalification Programs, the Quality Assurance Programs, and the Emergency Plans will not be materially changed by the proposed name change. The name change amendments will not affect the executive oversight provided by the Chief Nuclear Officer and his staff.

Therefore, the proposed amendments do not involve any increase in the probability or consequences of an accident previously analyzed.

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

Response: No.

The proposed amendments do not involve any change in the design, configuration, or operation of the nuclear plant. The current plant design, design bases, and plant safety analysis will remain the same.

The Limiting Conditions for Operations, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications are not affected by the proposed changes. As such, the plant conditions for which the design basis accident analyses were performed remain valid.

The proposed amendments do not introduce a new mode of plant operation or new accident precursors, do not involve any physical alterations to plant configurations, or make changes to system setpoints that could initiate a new or different kind of accident.

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

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

Response: No.

The proposed amendments do not involve a change in the design, configuration, or operation of the nuclear plants. The change does not affect either the way in which the plant structures, systems, and components perform their safety function or their design and licensing bases.

Plant safety margins are established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications. Because there is no change to the physical design of the plant, there is no change to any of these margins.

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

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: November 7, 2007.

Description of amendment request: The proposed amendment would delete License Condition 2.F, which requires reporting of violations of certain other requirements contained in Section 2.C of the license.

The NRC staff issued a ''Notice of Availability of Model Application Concerning Elimination of Typical License Condition Requiring Reporting of Violations of Section 2.C of Operating License Using the Consolidated Line Item Improvement Process'' in the **Federal Register** on November 4, 2005 (70 FR 67202). The notice referenced a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request published in the **Federal Register** on August 29, 2005 (70 FR 51098). In its application dated November 7, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

Response: No.

The proposed change involves the deletion of a reporting requirement. The change does not affect plant equipment or operating practices and therefore does not significantly increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change is administrative in that it deletes a reporting requirement. The change does not add new plant equipment, change existing plant equipment, or affect the operating practices of the facility. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change deletes a reporting requirement. The change does not affect plant equipment or operating practices and therefore does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis adopted by the licensee and, based upon this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendment involves NSHC.

Attorney for licensee: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Entergy Nuclear Operations, Inc., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2 (IP2), Westchester County, New York

Date of amendment request: October 24, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) requirements related to the containment buffering agent used for pH control under post loss-of-coolant accident (LOCA) conditions. Specifically, the proposal would approve the use of sodium tetraborate (STB) as the buffering agent instead of the currently approved compound, trisodium phosphate (TSP). The reason for this change in buffering agents is to minimize the potential for an adverse chemical interaction between the TSP and certain insulation materials in the containment that could degrade flow through the sump screens following certain design-basis accident scenarios such as a LOCA.

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 amendment does not involve a significant increase in the probability of an accident previously evaluated because the containment buffering agent is not an initiator of any analyzed accident. The proposed change does not impact any failure modes that could lead to an accident.

The proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated. The buffering agent in containment is designed to buffer the acids expected to be produced after a LOCA and is credited in the radiological analysis for iodine retention. Utilizing STB as a buffering agent ensures the post LOCA containment sump mixture will have a pH $\geq$ 7.0. The proposed change of replacing TSP with STB results in the radiological consequences remaining within the limits of 10 CFR 50.67 as demonstrated by existing analyses of record.

Therefore, operation of the facility in accordance with the proposed amendment would 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 amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. STB is a passive component that is proposed to be used at IP2 as a buffering agent to increase the pH of the initially acidic post-LOCA containment water to a more neutral pH. Changing the proposed buffering agent from TSP to STB does not constitute an accident initiator or create a new or different kind of accident previously analyzed. The proposed amendment does not involve operation of any required systems, structures or components in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the changes being requested.

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

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

Response—No.

The proposed amendment does not involve a significant reduction in a margin of safety. The proposed amendment of changing the buffering agent from TSP to STB results in equivalent control of maintaining sump pH at 7.0 or greater, thereby controlling containment atmosphere iodine and ensuring the radiological consequences of a LOCA are within regulatory limits. The use of STB also reduces the potential for exacerbating sump screen blockage due to a chemical interaction between TSP and certain calcium sources used in containment. This proposed amendment eliminates the formation of calcium phosphate precipitate thereby reducing the overall amount of precipitate that may be formed in a postulated LOCA. The buffer change would minimize the potential chemical effects and should enhance the ability of the emergency core cooling system to perform the post-accident mitigating functions.

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

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

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

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: October 24, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) requirements regarding the setpoint and definition of the low-low level alarm on the Refueling Water Storage Tank (RWST). Specifically, the proposal would revise the setpoint of the low-low level alarm from a range of greater than or equal to 10.5 ft and less than or equal to 12.5 ft to a range of greater than or equal to 9.0 ft and less than or equal to 11.0 ft, and revise the definition of the RWST ''low level alarm'' to ''low-low level alarm.'' The reason for these changes is to ensure that adequate water is supplied to the containment floor to eliminate the risk of vortexing and/or draw down at the sump strainer modules following a small-break loss-of-coolant accident (LOCA). The proposed changes are being requested to support resolution of the pressurized-water reactor sump performance issue involving debris accumulation, Generic Safety Issue (GSI)–191.

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

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

Response: No.

The proposed changes to the Technical Specifications are consistent with the assumptions of all design basis accidents, as they exist currently and as affected subsequent to the implementation of the proposed amendment. The change in the RWST low-low level alarm setpoint range has been demonstrated to be within the safety margins for post-accident parameters and, in most cases, actually beneficial to plant post-accident response capability. The RWST is designed to respond to a variety of accidents, and, for operation in Modes 1 through 4, it serves no other purpose. Therefore, any adjustment of an intermediate level setpoint cannot increase the probability of a design basis accident. The change in the definition of the RWST ''low level alarm'' to ''low-low level alarm'' is editorial and therefore does not affect the function of the alarm. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes represent a minor adjustment to an existing setpoint range. The effect of the changes will be to assure recirculation flow following a LOCA with consideration for sump strainer installation, in response to GSI–191. However, the RWST will continue to perform its function in essentially the same manner that it has since original plant design. No changes in equipment operation or procedural control will result from this amendment that could possibly degrade the performance of the RWST or cause it to be operated in a manner inconsistent with existing design basis assumptions. The change in the definition of the RWST ''low level alarm'' to ''low-low level alarm'' is editorial and therefore does not affect the function of the alarm. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed changes improve the margin to safety, especially with respect to post-accident temperature/pressure and dose consequences during injection and, most importantly, pump performance under postulated sump debris conditions during recirculation. Significant margin is available to preclude air ingestion in the ECCS [emergency core cooling system] pumps, and sufficient time is available for the operators to perform the switchover to recirculation. The change in the definition of the RWST ''low level alarm'' to ''low-low level alarm'' is editorial and therefore does not affect the function of the alarm. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Mark G. Kowal.

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

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

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

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

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

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

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

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: July 19, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Sections 5.3.1/6.3.1, ''Unit Staff Qualifications,'' for operator license applicants in accordance with current industry standards for education and experience eligibility requirements. The proposed amendment would permit changes to the unit staff qualification education and experience eligibility requirements for licensed operators. The proposal will bring Exelon Generation Company, LLC (EGC) and AmerGen Energy Company, LLC (AmerGen) sites in alignment with current industry practices and facilitate the development of a pre-initial licensed operator training 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 the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Licensed operator qualification and training can have an indirect impact on accidents previously evaluated. However, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, determined that this impact remains acceptable when licensees have an accredited licensed operator training program which is based on a systems approach to training (SAT). The NRC has concluded in RIS [Regulatory Issue Summary] 2001–01 and NUREG–1021 that standards and guidelines applied by INPO [the Institute of Nuclear Power Operations] in their accredited training programs are equivalent to those put forth by or endorsed by the NRC. Therefore, maintaining an INPO accredited SAT licensed operator training program is equivalent to maintaining an NRC approved licensed operator training program which conforms with applicable NRC Regulatory Guidelines or NRC endorsed industry standards. The proposed changes conform to ACAD [air containment atmosphere distribution] 00–003, Revision 1 licensed operator education and experience eligibility requirements.

Based on the above, EGC and AmerGen conclude that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment involves changes to the licensed operator training programs, which are administrative in nature. The EGC and AmerGen licensed operator training programs have been accredited by INPO and are based on SAT.

Based on the above discussion, EGC and AmerGen conclude that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed TS changes are administrative in nature. The proposed TS changes do not affect plant design, hardware, system operation, or procedures for accident mitigation systems. The proposed changes do not impact the performance or proficiency requirements for licensed operators. As a result, the ability of the plant to respond to and mitigate accidents is unchanged by the proposed TS changes. Therefore, these changes do not involve a significant reduction in a margin of safety.

Based on the above, EGC and AmerGen conclude that the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above evaluation of the three criteria, EGC and AmerGen conclude that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Branch Chief: Russell Gibbs.

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

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

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

Exelon Generation Company, LLC, Docket No. 50–352 and No. 50–353, Limerick Generating Station, Unit 1 and 2, Montgomery County, Pennsylvania

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3,York and Lancaster Counties, Pennsylvania

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

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: August 8, 2007.

Description of amendment request: The proposed amendment replaces references to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in the applicable technical specification (TS) section for the Inservice Testing Program (IST) for the Exelon Generation Company, LLC, and AmerGen Energy Company, LLC, (the licensees) plants that have implemented industry Improved Technical Specifications. The proposed changes are based on TS Task Force (TSTF) 479–A, Revision 0, ''Changes to Reflect Revision of 10 CFR 50.55a,'' as modified by TSTF–497, Revision 0, ''Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less.'' In addition, the proposed amendment adds a provision in the applicable TS section to only apply the extension allowance of SR 3.0.2 to the frequency table listed in the TS as part of the IST and to normal and accelerated inservice testing frequencies of two years or less, as applicable.

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

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

Response: No.

The proposed changes revise the applicable TS Section to conform to the requirements of 10 CFR 50.55a, ''Codes and standards,'' paragraph (f) regarding the inservice testing of pumps and valves. The current TS reference the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code as applicable, which is consistent with 10 CFR 50 .55a, paragraph (f), ''Inservice testing requirements.'' In addition, the proposed changes clarify that the extension allowance of SR 3.0.2 only applies to the frequency table listed in the TS, if applicable, as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The definitions of the frequencies are not changed by this license amendment request.

The proposed changes are administrative in nature, do not affect any accident initiators, do not affect the ability to successfully respond to previously evaluated accidents and do not affect radiological assumptions used in the evaluations. Thus, the probability or radiological consequences of any accident previously evaluated are not increased.

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

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

Response: No.

The proposed changes revise the applicable TS Section to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. The current TS Section references the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code as applicable, which is consistent with 10 CFR 50.55a(f). In addition, the proposed changes clarify that the extension allowance of SR 3.0.2 only applies to the frequency table listed in the TS, if applicable, as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The definitions of the frequencies are not changed by this license amendment request.

The proposed changes to the applicable TS Section do not affect the performance of any structure, system, or component credited with mitigating any accident previously evaluated and do not introduce any new modes of system operation or failure mechanisms.

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

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

Response: No.

The proposed changes revise the applicable TS Section for Braidwood Station Units 1 and 2, Byron Station Units 1 and 2, Dresden Nuclear Power Station Units 2 and 3, Limerick Generating Station Units 1 and 2, Oyster Creek Generating Station, Peach Bottom Atomic Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, and Three Mile Island Unit 1 to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves.

The current TS Section references the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code as applicable, which is consistent with the 10 CFR 50.55a(f). In addition, the proposed changes clarify that the extension allowance of SR 3.0.2 only applies to the frequency table listed in the TS, if applicable, as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The definitions of the frequencies are not changed by this license amendment request.

The proposed changes do not modify the safety limits or setpoints at which protective actions are initiated and do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

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

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

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

NRC Branch Chief: Russell Gibbs.

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 amendment request: August 1, 2007.

Description of amendment request: The proposed amendment would revise the technical specification (TS) allowable value (AV) for the Reactor Protection System (RPS) Instrumentation Function 10, ''Turbine Condenser Vacuum—Low,'' specified in TS Table 3.3.1.1–1, ''Reactor Protection System Instrumentation.'' The proposed amendment also revises the Channel Functional Test (CFT) and Channel Calibration (CC) Surveillance Test Interval (STI) for DNPS TS Table 3.3.1.1–1, Function 10. As part of the DNPS STI revision, surveillance requirement (SR) 3.3.1.10, ''Channel Calibration,'' which is specific to the Turbine Condenser Vacuum—Low instrument function, is deleted since it is no longer applicable.

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

Revision of Allowable Value

The proposed license amendment implements a revised AV for the Turbine Condenser Vacuum—Low scram instrument function at DNPS, Units 2 and 3 and QCNPS, Units 1 and 2.

The proposed changes to the DNPS and QCNPS Turbine Condenser Vacuum—Low scram AV do not require modification [of] any system interface or affect the probability of any event initiators at the facilities. Overall RPS performance will remain within the bounds of the previously performed accident analyses, since no hardware changes are proposed.

There will be no degradation in the performance of, or an increase in the number of challenges imposed on safety-related equipment that are assumed 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 Updated Final Safety Analysis Report. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

For these reasons, the proposed DNPS and QCNPS AV changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Relaxation of STIs (DNPS only)

The proposed license amendment implements a revised CFT and CC STI for the Turbine Condenser Vacuum—Low scram instrument function at DNPS Units 2 and 3. The proposed DNPS TS change to increase the CFT STI for the Turbine Condenser Vacuum—Low scram instrument function is based on an analytical method that has been reviewed and approved by the NRC [Nuclear Regulatory Commission].

The proposed change to relax the CFT STI implements recommendations from a generic evaluation that was developed by General Electric (GE) and the Boiling Water Reactor Owners' Group (BWROG), and subsequently approved by the NRC. This licensing topical report (LTR) assessed the reliability of TS actuation instrumentation and concluded that extending AOTS [allowed outage times] and CFT STIs for test and repair activities would enhance operational safety.

The proposed DNPS TS change to increase the CC STI for the Turbine Condenser Vacuum—Low scram instrument function is based upon a revised setpoint error analysis that provides revised AVs, trip setpoints, and Expanded Tolerances (ETs) for the instrument. These new AVs, trip setpoints, and ETs establish increased design margin between the nominal trip setpoint and the AV. This increased design margin, combined with historical CC data, provides adequate assurance that the component will remain operable when necessary for the prevention or mitigation of accidents or transients.

The TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these proposed STI changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed changes do not involve any physical changes to plant systems, structures or components (SSCs), or the manner in which these SSCs are operated. These changes will not alter the operation of equipment assumed to be available for the mitigation of anticipated operational occurrences (AOOs) by the plant safety analysis or licensing basis.

The proposed deletion of SR 3.3.1.10 is an administrative change, since the SR will no longer be applicable to any instrument function in DNPS TS Table 3.3.1–1. Therefore, the proposed deletion of SR 3.3.1.10 will not impact the testing, calibration, and inspection of RPS instrumentation that is necessary to assure that the quality of the instrumentation is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

For these reasons, the proposed DNPS STI changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

In summary, the proposed license changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes to the DNPS and QCNPS Turbine Condenser Vacuum—Low scram AV and the DNPS CFT and CC STIs do not affect the design, functional performance, or operation of the facility. Similarly, the proposed changes do not affect the design or operation of any SSCs involved in the mitigation of any accidents, nor do they affect the design or operation of any component in the facilities such that new equipment failure modes are created.

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

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

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

The proposed DNPS and QCNPS AV change does not affect the acceptance criteria for any analyzed event, nor is there a change to any Safety Analysis Limit. There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. All required design functions are maintained, and the AVs, are consistent with NRC-approved methodology and guidance for establishment of TS AVs.

The proposed AV changes do not affect the accident analyses that assume operability of the instrument associated with the AV. The Turbine Condenser Vacuum—Low scram function is credited in the Loss of Main Condenser Vacuum AOO. The loss of main condenser AOO event assumes that the main condenser is instantaneously lost while the unit is operating at full power. This is classified as a moderate frequency event and is described in the UFSAR [updated final safety analysis report] as being bounded by the turbine trip with bypass failure event.

The worst case for this AOO would occur if the loss of vacuum were instantaneous. In this case, the loss of main condenser event would be identical to the turbine trip with bypass failure event. During a turbine trip with bypass failure event, the primary system relief valves would remove the majority of the stored heat, while the IC [isolation condenser] at DNPS and RCIC [reactor core isolation cooling] at QCNPS would remove the remaining decay heat. Slower losses of condenser vacuum would produce less severe AOOs, since the turbine stop valves and bypass valves will still be available prior to vacuum levels reaching the nominal trip setpoint for the turbine trip and turbine bypass valve closure scram.

In that the proposed reduction of the Turbine Condenser Vacuum—Low AV is based upon an AL [analytical limit] that is equal to the nominal trip setpoint for the turbine trip, the resulting nominal trip setpoint for the Turbine Condenser Vacuum—Low scram will still be more conservative than the turbine trip setpoint. Therefore, the sequence of events for the loss of main condenser AOO will still result in a reactor scram prior to the turbine trip. Since the proposed change to the Turbine Condenser Vacuum—Low AV will not impact the limiting AOO analysis (i.e., the turbine trip with bypass failure event), the proposed change does not reduce any margin of safety.

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

The proposed DNPS CFT STI change is based on an NRC-approved generic analysis. This analysis concluded that the proposed CFT STI change does not significantly affect the probability of failure or availability of the affected instrumentation systems. Therefore, the proposed DNPS CFT STI change does not affect the accident analyses that assume operability of the instrument associated with the AV.

The proposed DNPS CC STI change is based on a revised setpoint error analysis for the Turbine Condenser Vacuum—Low scram instrument function that provides a revised AV, trip setpoint, and Expanded Tolerance (ET) for the instrument. The new AV, trip setpoint, and ET establish increased design margin between the nominal trip setpoint and the AV. This increased design margin, combined with historical CC data, provides adequate assurance that the component will remain operable when necessary for the prevention or mitigation of accidents or transients. Therefore, the proposed DNPS CFT STI change does not affect the accident analyses that assume operability of the instrument associated with the AV.

Therefore, the proposed changes to extend the DNPS CFT and CC STIs do not involve a significant reduction in the margin of safety.

In summary, the proposed DNPS and QCNPS AV changes and DNPS STI changes do not involve a significant reduction in the margin of safety.

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

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

NRC Branch Chief: Russell Gibbs.

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

Date of amendment request: October 9, 2007.

Description of amendment request: The proposed amendment would change the technical specifications (TS) of Dresden Nuclear Power Station (DNPS), Units 2 and 3, consistent with TS Task Force (TSTF) Change Traveler TSTF–423 to the standard TSs boiling water reactor plants, to allow, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Changes proposed herein will be made to the DNPS, Units 2 and 3, TSs for selected Required Action end states providing this allowance.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 14, 2005 (70 FR 74037), on possible license amendments adopting TSTF–423 using the NRC's consolidated line item improvement process (CLIIP) for amending licensee's TSs, which included a model safety evaluation (SE) and model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 26, 2006 (71 FR 14726), which included the resolution of public comments on the model SE. The licensee affirmed the applicability of the following NSHC determination in its application.

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

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

The proposed change allows a change to certain required end states when the TS Completion Times for remaining in power operation will be exceeded. Most of the requested technical specification (TS) changes are to permit an end state of hot shutdown (Mode 3) rather than an end state of cold shutdown (Mode 4) contained in the current TS. The request was limited to: (1) Those end states where entry into the shutdown mode is for a short interval, (2) entry is initiated by inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable technical specification, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical. Risk insights from both the qualitative and quantitative risk assessments were used in specific TS assessments. Such assessments are documented in Section 6 of GE NEDC–32988, Revision 2, ''Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR Plants.'' They provide an integrated discussion of deterministic and probabilistic issues, focusing on specific technical specifications, which are used to support the proposed TS end state and associated restrictions. The staff finds that the risk insights support the conclusions of the specific TS assessments. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident after adopting proposed TSTF–423, are no different than the consequences of an accident prior to adopting TSTF–423. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). If risk is assessed and managed, allowing a change to certain required end states when the TS Completion Times for remaining in power operation are exceeded, i.e., entry into hot shutdown rather than cold shutdown to repair equipment, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change and the commitment by the licensee to adhere to the guidance in TSTF–IG–05–02, Implementation Guidance for TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A,'' will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed. The BWROG's risk assessment approach is comprehensive and follows staff guidance as documented in RGs 1.174 and 1.177. In addition, the analyses show that the criteria of the three-tiered approach for allowing TS changes are met. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A risk assessment was performed to justify the proposed TS changes. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Russell Gibbs.

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

Date of amendment request: September 14, 2007.

Description of amendment request: Duane Arnold Energy Center requests a proposed change to plant specific technical specifications (TS) 3.3.2.1, ''Control Rod Block Instrumentation,'' to allow the use of the improved Banked Position Withdrawal Sequence (BPWS) during shutdowns in accordance with NEDO–33091–A, Revision 2, ''Improved BPWS Control Rod Insertion Process,'' dated July 2004. The proposed changes are consistent with Nuclear Regulatory Commission (NRC)-approved Industry Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–476, Revision 1, ''Improved BPWS Control Rod Insertion Process (NEDO–33091).''

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

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

The proposed changes modify the TS to allow the use of the improved banked position withdrawal sequence (BPWS) during shutdowns if the conditions of NEDO–33091–A, Revision 2, ''Improved BPWS Control Rod Insertion Process,'' July 2004, have been satisfied. The staff finds that the licensee's justifications to support the specific TS changes are consistent with the approved topical report and TSTF–476, Revision 1. Since the change only involves changes in control rod sequencing, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident after adopting TSTF–476 are no different than the consequences of an accident prior to adopting TSTF–476. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The control rod drop accident (CRDA) is the design basis accident for the subject TS changes. This change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change, TSTF–476, Revision 1, incorporates the improved BPWS, previously approved in NEDO–33091–A, into the improved TS. The control rod drop accident (CRDA) is the design basis accident for the subject TS changes. In order to minimize the impact of a CRDA, the BPWS process was developed to minimize control rod reactivity worth for BWR plants. The proposed improved BPWS further simplifies the control rod insertion process, and in order to evaluate it, the staff followed the guidelines of Standard Review Plan Section 15.4.9, and referred to General Design Criterion 28 of Appendix A to 10 CFR Part 50 as its regulatory requirement. The TSTF stated the improved BPWS provides the following benefits: (1) Allows the plant to reach the all-rods-in condition prior to significant reactor cool down, which reduces the potential for re-criticality as the reactor cools down; (2) reduces the potential for an operator reactivity control error by reducing the total number of control rod manipulations; (3) minimizes the need for manual scrams during plant shutdowns, resulting in less wear on control rod drive (CRD) system components and CRD mechanisms; and, (4) eliminates unnecessary control rod manipulations at low power, resulting in less wear on reactor manual control and CRD system components. The addition of procedural requirements and verifications specified in NEDO–33091–A, along with the proper use of the BPWS will prevent a control rod drop accident (CRDA) from occurring while power is below the low power setpoint (LPSP). The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Attorney for licensee: Marjan Mashhadi, Florida Power & Light Company, 801 Pennsylvania Avenue, Suite 220, Washington, DC 20004.

NRC Acting Branch Chief: Clifford G. Munson.

FPL Energy Point Beach, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: October 12, 2007.

Description of amendment request: FPL Energy Point Beach, LLC (FPLE–PB) proposes to revise Technical Specification (TS) 5.5.1 5 ''Containment Leakage Rate Testing Program,'' for Units 1 and 2. The proposed change would allow a one-time interval extension of no more than 5 years for the Type A, Integrated Leakage Rate Test (ILRT).

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.

This license amendment proposes to revise the Technical Specifications (TS) to allow for the one-time extension of the containment integrated leakage rate test interval from 10 to 15 years. The containment vessel function is to mitigate consequences of an accident. There are no design basis accidents initiated by a failure of the containment leakage mitigation function. The extension of the containment integrated leakage rate test interval will not create an adverse interaction with other systems that could result in initiation of a design basis accident. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the containment integrated leakage rate test interval from 10 to 15 years. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be of a magnitude that NUREG–1493 indicates is very small. FPLE–PB has also analyzed the increase in risk in terms of the frequency of large early releases from accidents. The increase in the large early release frequency resulting from the proposed extension was determined to be within the guidelines published in RG 1.I74. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. FPLE–PB has determined that the increase in conditional containment failure probability from reducing the containment integrated leakage rate test frequency from one test per 10 years to one test per 15 years would be small.

Continued containment integrity is also assured by the history of successful containment integrated leakage rate tests, and the established programs for local leakage rate testing and IWE inservice inspections which are not affected by the proposed change. Therefore, the probability of occurrence or the consequences of an accident previously analyzed are not significantly increased.

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

Response: No.

The proposed change to extend the containment integrated leakage rate test interval from 10 to 15 years does not create any new or different accident initiators or precursors. The length of the containment integrated leakage rate test interval does not affect the manner in which any accident begins. The proposed change does not create any new failure modes for the containment and does not affect the interaction between the containment and any other system. Thus, the proposed changes do 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 risk-based margins of safety associated with the containment integrated leakage rate test are those associated with the estimated person-rem per year, the large early release frequency and the conditional containment failure probability. FPLE–PB has quantified the potential effect of the proposed change on these parameters and determined that the effect is not significant. The non-risk-based margins of safety associated with the containment integrated leakage rate test are those involved with its structural integrity and leak tightness. The proposed change to extend the containment integrated leakage rate test interval from 10 to 15 years does not adversely affect either of these attributes. The proposed change only affects the frequency at which these attributes are verified. Therefore, the proposed change does not involve a significant reduction in 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. Antonio Fernandez, Senior Attorney, FPL Energy, LLC, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Acting Branch Chief: Cliff Munson.

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

Date of amendment request: October 22, 2007.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 3.6.3.1, Primary Containment Hydrogen Recombiners, and references to the hydrogen and oxygen monitors in TS 3.3.3.1, Post Accident Monitoring (PAM) Instrumentation. The proposed TS changes support implementation of the revisions to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.44, ''Combustible gas control for nuclear power reactors,'' that became effective on October 16, 2003. These changes are consistent with Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.'' The availability of this TS improvement was announced in the **Federal Register** on September 25, 2003 (68 FR 55416) as part of the consolidated line item improvement process. The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

The proposed amendment would also relocate, from the Renewed Facility Operating License to the NMP2 Updated Safety Analysis Report, License paragraph 2.C.(11a), *Additional Condition 3,* which requires establishing containment hydrogen monitoring within 90 minutes of initiating emergency core cooling following a loss-of-coolant accident.

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:

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

The revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen [and oxygen] monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1 is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen [and oxygen] monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. [Also, as part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.]

The regulatory requirements for the hydrogen [and oxygen] monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, [classification of the oxygen monitors as Category 2] and removal of the hydrogen [and oxygen] monitors from TS will not prevent an accident management strategy through the use of the SAMGs [severe accident management guidelines], the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen [and oxygen] monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen [and oxygen] monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen [and oxygen] monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen [and oxygen] monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in [a] Margin of Safety

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen [and oxygen] monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

[Category 2 oxygen monitors are adequate to verify the status of an inerted containment.]

Therefore, this change does not involve a significant reduction in [a] margin of safety. [The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors.] Removal of hydrogen [and oxygen] monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

The NRC staff has reviewed the analysis adopted by the licensee and, based on this review, it appears that the 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

PSEG Nuclear LLC, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: October 17, 2007.

Description of amendment request: The proposed amendment would allow a one-time revision to the requirements for fuel decay time prior to commencing movement of irradiated fuel in the reactor pressure vessel (RPV). Currently, Technical Specification (TS) 3/4.9.3, ''Decay Time'' requires that: (a) The reactor has been subcritical for at least 100 hours prior to movement of irradiated fuel in the RPV between October 15th through May 15th; and (b) the reactor has been subcritical for at least 168 hours prior to movement of irradiated fuel in the RPV between May 16th and October 14th. The calendar approach is based on average river water temperature which is cooler in the fall through spring months. The proposed amendment would revise TS 3/4.9.3 to allow fuel movement to commence at 86 hours after the reactor is subcritical. The proposed change would only be applicable to Salem Nuclear Generating Station, Unit No. 2 refueling outage 2R16, which is scheduled to commence on March 4, 2008.

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

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

Response: No.

The proposed license amendment would allow fuel assemblies to be removed from the reactor core and be stored in the Spent Fuel Pool [SFP] in less time after subcriticality than currently allowed by the TSs. Decreasing the decay time of the fuel affects the radionuclide make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The accident previously evaluated that is associated with the proposed license amendment is the fuel handling accident [FHA]. Allowing the fuel to be offloaded in less time after subcriticality using actual heat loads does not impact the manner in which the fuel is offloaded. The accident initiator is the dropping of the fuel assembly. Since earlier offload does not affect fuel handling, there is no increase in the probability of occurrence of a [FHA]. The time frame in which the fuel assemblies are moved has been evaluated against the [Title 10 of the Code of Federal Regulations (10 CFR) Section 50.67] dose limits for members of the public, licensee personnel and control room. Additionally, the guidance provided in [Regulatory Guide (RG)] 1.183 was used for the selective application of Alternative Source Term. All dose limits are met with the reduced core offload times; and significant margin is maintained, as the minimum decay time prior to movement of fuel for the FHA analysis is 24 hours.

Therefore, the proposed license amendment does not significantly increase the probability [ ] or the consequences of accidents previously evaluated.

2. [Does the change] [c]reate the possibility of a new or different kind of accident from any accident previously evaluated[?]

Response: No.

The proposed license amendment would allow core offload to occur in less time after subcriticality which affects the radionuclide makeup of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The radionuclide makeup of the fuel assemblies and the amount of decay heat produced by the fuel assemblies do not currently initiate any accident. A change in the radionuclide makeup of the fuel at the time of core offload or an increase in the decay heat produced by the fuel being offloaded will not cause the initiation of any accident. The accident previously evaluated that is associated with fuel movement is the [FHA]; no new accidents are introduced. There is no change to the manner in which fuel is being handled or in the equipment used to offload or store the fuel. The effects of the additional decay heat load have been analyzed. The analysis demonstrates that the existing [SFP] cooling system and associated systems under worst-case circumstances would maintain licensing limits and the integrity of the [SFP].

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

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

Response: No.

The margin of safety pertinent to the proposed changes is the dose consequences resulting from a [FHA]. The shorter decay time prior to fuel movement has been evaluated against 10 CFR 50.67 and all limits continue to be met. All dose limits are met with the reduced core offload times; and significant margin is maintained, as the minimum decay time prior to movement of fuel for the FHA analysis is 24 hours. Decay heat-up calculations performed prior to the refueling outage as part of the IDHM [Integrated Decay Heat Management] program ensure that planned spent fuel transfer to the SFP will not result in maximum SFP temperature exceeding the design basis limit of 149°F (with both heat exchangers available) or 180°F (with one heat exchanger alternating between the two pools). As stated above, the changes in radionuclide makeup and additional heat load do not impact any safety settings and do not cause any safety limit to not be met. In addition, the integrity of the [SFP] is maintained.

The time frame in which the fuel assemblies are moved has been evaluated against the 10 CFR 50.67 dose limits for members of the public, licensee personnel and control room. Additionally, the guidance provided in [RG] 1.183 was used. Calculations performed conclude that expected dose limits following a [FHA] are met with the proposed decay time prior to commencing fuel movement.

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

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: October 27, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to establish more effective and appropriate action, surveillance, and administrative requirements related to ensuring the habitability of the control room envelope (CRE) in accordance with Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification change traveler TSTF–448, Revision 3, ''Control Room Habitability.'' Specifically, the proposed amendment would modify TS 3.7.7, ''Control Room Emergency Ventilation System,'' and TS Section 6, ''Administrative Controls.'' The NRC staff issued a ''Notice of Availability of Technical Specification Improvement to Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process associated with TSTF–448, Revision 3, in the **Federal Register** on January 17, 2007 (72 FR 2022). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated October 27, 2007, Tennessee Valley Authority (the licensee) affirmed the applicability of the model NSHC determination which is presented below.

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

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

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does 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 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas H. Boyce.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: February 16, 2007.

Brief description of amendment request: The proposed amendment would revise Technical Specification 3/4.4.3, ''Reactor Coolant System, Relief Valves'' to modify the method of testing the pressurizer Power Operated Relief Valves (PORVs). Specifically the requirement for bench testing the valves is changed to accommodate testing of the PORVs while installed in the plant. The change is requested due to the installation of new PORVs that are welded to the piping rather than bolted into the system.

Date of publication of individual notice in Federal Register: November 19, 2007.

Expiration date of individual notice: December 19, 2007 (public comment), January 18, 2008 (hearing requests).

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

Duke Power Company LLC, et. al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of application for amendments: April 30, 2007.

Brief description of amendments: The amendment revised Technical Specification (TS) 5.5.9, ''team Generator (SG) Tube Surveillance Program,'' regarding the required SG inspection scope for Catawba Unit 2 during the End of Cycle 15 Refueling Outage and Operating Cycle 16. The changes modified the tube repair criteria for portions of the SG tubes within the hot leg tubesheet region of the SGs.

Date of issuance: October 31, 2007.

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

Amendment Nos.: 233.

Renewed Facility Operating License No. NPF–52: Amendments revised the licenses and the technical specifications.

Date of initial notice in Federal Register: August 13, 2007 (72 FR 45272).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2007.

No significant hazards consideration comments received: No.

Duke Power Company LLC, et. al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: March 29, 2007, as supplemented September 7, 2007, October 9 and October 12, 2007.

Brief description of amendments: The amendments revised the Catawba 1 and 2, Technical Specifications 3.5.2.8, and authorized changes to the updated final safety analysis report concerning modifications to the emergency core cooling system sump.

Date of issuance: November 8, 2007.

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

Amendment Nos.: 238, 234.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the licenses and the technical specifications.

Date of initial notice in Federal Register: August 13, 2007 (72 FR 45274). The supplements dated September 7, 2007, October 9, and October 12, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 8, 2007.

No significant hazards consideration comments received: No.

Duke Power Company LLC, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: November 16, 2006, supplemented May 9 and August 28, 2007.

Brief description of amendments: The amendments authorized revision of the Updated Final Safety Analysis Report to describe the flood protection measures for the auxiliary building.

Date of Issuance: November 14, 2007.

Effective date: As of the date of issuance and shall be implemented within 30 days after completion of the flood protection measures for the auxiliary building.

Amendment Nos.: 357, 359, and 358.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the licenses.

Date of initial notice in Federal Register: January 3, 2007 (72 FR 151). The supplements dated May 9 and August 28, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 14, 2007.

No significant hazards consideration comments received: No.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: July 16, 2007, as supplemented by letter dated August 7, 2007.

Brief description of amendment: The proposed amendment revised the facility operating license (FOL), Paragraph 2.C, and technical specifications (TS) 3.7.2 and TS 5.5 for River Bend Station, Unit 1.

Date of issuance: November 16, 2007.

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

Amendment No.: 154.

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

Date of initial notice in Federal Register: September 11, 2007 (72 FR 51857). The supplement dated August 7, 2007, 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** on September 11, 2007 (72 FR 51857). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 16, 2007.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendment: November 27, 2006, as supplemented by letter dated August 24, 2007.

Brief description of amendment: This amendment revises multiple TSs relating to testing of the Emergency Diesel Generators (EDGs). Specifically, the changes eliminate various accelerated testing requirements, eliminate the EDG test schedule table based on failure rates, relax acceptance criteria associated with the ''fast start'' and load rejection tests and eliminate the EDG failure report.

Date of issuance: November 6, 2007.

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

Amendment Nos.: 189 and 150.

Facility Operating License Nos. NPF–39 and NPF–85: This amendment revised the license and Technical Specifications.

Date of initial notice in Federal Register: July 31, 2007 (72 FR 41784).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 6, 2007.

No significant hazards consideration comments received: No.

FPL Energy Point Beach, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: June 29, 2007.

Brief description of amendments: The amendments would modify the Technical Specifications (TSs) 3.7.2, by removing the specific isolation time for the main steam isolation valves from the associated TS surveillance requirements and by replacing it with the requirement to verify the valve isolation time is within limits.

Date of issuance: November 16, 2007.

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

Amendment Nos.: 230, 235.

Renewed Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications/ License.

Date of initial notice in Federal Register: September 11, 2007 (72 FR 51865).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 16, 2007.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: July 9, 2007.

Brief description of amendment: The amendment revised the Technical Specifications by removing the Table of Contents.

Date of issuance: November 8, 2007.

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

Amendment No.: 152.

Facility Operating License No. DPR–22.

Amendment revised the Technical Specifications. Date of initial notice in Federal Register: August 14, 2007 (72 FR 45459).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 8, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 15, 2007, as supplemented on September 6, 2007.

Brief description of amendments: The amendments revise the licensing basis, as described in Appendix 3A of the Salem Updated Final Safety Analysis Report (UFSAR), regarding the method of calculating the net positive suction head available for the emergency core cooling system and containment heat removal system pumps. These changes to the Salem licensing basis relate to issues associated with Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors.''

Date of issuance: November 15, 2007.

Effective date: As of the date of issuance, to be implemented by December 31, 2007.

Amendment Nos.: 285 and 268.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revise the UFSAR.

Date of initial notice in Federal Register: September 11, 2007 (72 FR 51866). The letter dated September 6, 2007, 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 November 15, 2007.

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 e-mail to *pdr@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, person(s) may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request via electronic submission through the NRC E-Filing system for a hearing and a petition for leave to intervene. 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@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 petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner 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 petitioner/requestor 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 petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the authority to act for the petitioners/requestors with respect to that contention.

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.

A request for hearing or a petition for leave to intervene must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated in August 28, 2007, (72 FR 49139). The E-Filing process requires participants to submit and serve documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *HEARINGDOCKET@NRC.GOV,* or by calling (301) 415–1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to

[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.

download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at *http://www.nrc.gov/site-help/e-submittals/install-viewer.html.* Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.*

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The EIE system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the ''Contact Us'' link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html* or by calling the NRC technical help line, which is available between 8:30 a.m. and 4:15 p.m., Eastern Time, Monday through Friday. The help line number is (800) 397–4209 or locally, (301) 415–4737.

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

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)-(viii). To be timely, filings must be submitted no later than 11:59 p.m. Eastern Time on the due date.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp*, unless excluded pursuant to an order of the Commission, an Atomic Safety and Licensing Board, or a Presiding Officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings. 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.

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

Date of application for amendments: October 22, 2007, as supplemented November 2 and November 9, 2007.

Brief Description of amendments: This amendment adds a new license condition, P.(3), to license Nos. DPR–32 and DPR–37, which authorize the licensee to modify the GOTHIC code as described in the Updated Final Safety Analysis Report (UFSAR) and update the UFSAR as required by 10 CFR 50.71(e).

Date of issuance: November 15, 2007.

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

Amendment Nos.: 256, 255.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments revise the licenses.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. The notice provided an opportunity to submit comments (by November 13, 2007) on the Commission(s proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing (by December 31, 2007), but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated November 15, 2007.

Attorney for licensee: Ms. Lillian M. Cuoco, Esq.

NRC Branch Chief: Evangelos C. Marinos.

Dated at Rockville, Maryland, this 23rd day of November, 2007.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E7–23225 Filed 12–3–07; 8:45 am]

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UNITED STATES POSTAL SERVICE BOARD OF GOVERNORS

Sunshine Act Meeting

DATE AND TIME: Monday, December 10, 2007, at 11 a.m. and Tuesday, December 11, 2007, at 8:30 a.m. and 10:30 a.m.

PLACE: Washington, DC, at U.S. Postal Service Headquarters, 475 L'Enfant Plaza, SW., in the Benjamin Franklin Room.

STATUS: December 10—11 a.m.—Closed; December 11—8:30 a.m.—Open; December 11—10:30 a.m.—Closed.

MATTERS TO BE CONSIDERED:

Monday, December 10 at 11 a.m. (Closed)

1. Strategic Issues.
2. Financial Update.
3. Product Pricing Update.
4. Global Business Pricing for Customized Agreements.
5. Postal Regulatory Commission Opinion and Recommended Decision in Negotiated Service Agreement with