Thursday, December 14, 2006

9:30 a.m. Meeting with Advisory Committee on Nuclear Waste (ACNW) (Public Meeting) (*Contact:* John Larkins, 301–415–7360).

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

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*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Michelle Schroll, (301) 415–1662.

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The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

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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 2, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–9110 Filed 11–3–06; 9:57 am]

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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 October 13, 2006, to October 26, 2006. The last biweekly notice was published on October 24, 2006 (71 FR 62306).

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, 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. 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 persons 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 a hearing or a petition for leave to intervene must be filed 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; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) e-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

For further details with respect to this 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.*

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

Date of amendments request: September 28, 2006.

Description of amendments request: The proposed amendments would revise certain Technical Specification (TS) requirements for mode change limitations in Limiting Condition for Operation 3.0.4 and Surveillance Requirement 3.0.4. This request is consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Traveler number TSTF–359, Revision 9, ''Increase Flexibility in Mode Restraints.'' In addition, the proposed amendments would correct TS Example 1.4–1, ''Surveillance Requirements,'' to accurately reflect the changes made by TSTF–359, which is consistent with NRC-approved TSTF–485, Revision 0.

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

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

Response: No.

The proposed change revises Section 1.4, Frequency, ''Example 1.4–1,'' to be consistent with Surveillance Requirement (SR) 3.0.4 and Limiting Condition for Operation (LCO) 3.0.4. This change is considered administrative in that it modifies the example to demonstrate the proper application of SR 3.0.4 and LCO 3.0.4. The requirements of SR 3.0.4 and LCO 3.0.4 are clear and are clearly explained in the associated Bases. As a result, modifying the example will not result in a change in usage of the Technical Specifications (TS). The proposed change does not adversely affect accident initiators or precursors, the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Therefore, this change is considered administrative and will have no effect on the probability or consequences of any accident previously evaluated.

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

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

Response: No.

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

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

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

Response: No.

The proposed change is administrative and will have no effect on the application of the Technical Specification requirements. Therefore, the margin of safety provided by the Technical Specification requirements is unchanged.

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

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

Attorney for licensee: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072–2034.

NRC Branch Chief: David Terao.

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

Date of amendment request: March 17, 2006.

Description of amendment request: The proposed amendment would revise Millstone Power Station, Unit No. 2 Technical Specification (TS) 3.4.4 to replace the existing maximum and minimum pressurizer water volume and water level limits with a maximum water level limit. The associated TS bases will be updated to address the proposed change.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not change the accident analysis of record, maintains the current maximum operating pressurizer level at its present value, 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 consequences of any analyzed accident since the change does not affect the function of any equipment credited for accident mitigation. Therefore, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the proposed change does not modify any plant equipment and there is no impact on the capability of existing equipment to perform its intended functions and no new failure modes are introduced by the proposed change, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety?

The proposed change maintains the current maximum operating pressurizer level at its present value, and the acceptance criterion for the maximum pressurizer level is unchanged. Since there are no changes, the proposed change does not involve a reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Harold K. Chernoff.

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: October 16, 2006.

Description of amendment request: The proposed change will add an NRC previously approved topical report to the analytical methods referenced in Technical Specification (TS) Section 5.6.5, ''Core Operating Limits Report (COLR).'' The current method of performing the loss-of-coolant accident (LOCA) analyses will be replaced by an updated method described in AREVA NP (formerly known as Framatome or Siemens) topical report, ''EXEM BWR–2000 [Boiling-Water Reactor—2000] ECCS [Emergency Core Cooling System] Evaluation Model.''

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.

Core operating limits are established each operating cycle in accordance with TS 3.2, ''Power Distribution'' and TS 5.6.5, ''Core Operating Limits Report (COLR)''. These core operating limits ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). In addition, the Average Planar Linear Heat Generation Rate (APLHGR) operating limits imposed by Technical Specification 3.2.1 also ensure that the peak cladding temperature (PCT) during the postulated design basis LOCA does not exceed the 2200 °F limit specified in 10 CFR 50.46. The APLHGR is a measure of the average linear heat generation rate of all the fuel rods in a fuel assembly at any axial location.

The methods used to determine the operating limits are those previously found acceptable by the NRC and listed in TS section 5.6.5.b. A change to TS section 5.6.5.b is requested to include an updated LOCA analysis method, EXEM BWR–2000. The updated method will be used to determine the APLHGR operating limits imposed by Technical Specification 3.2.1. EXEM BWR–2000 has been reviewed and approved by the NRC and is applicable to the RBS [River Bend Station] plant design and the AREVA NP fuel being used at RBS. The application of the LOCA analytical model will continue to ensure that the APLHGR operating limits are established to protect the fuel cladding integrity during normal operation, AOOs, and the design basis LOCA.

The requested TS changes concern the use of analytical methods and do not involve any plant modifications or operational changes that could affect any postulated accident precursors or accident mitigation systems and do not introduce any new accident initiation mechanisms.

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

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

Response: No.

The proposed TS amendment will not change the design function, reliability, performance, or operation of any plant systems, components, or structures. It does not create the possibility of a new failure mechanism, malfunction, or accident initiators not considered in the design and licensing bases. Plant operation will continue to be within the core operating limits that are established using NRC approved methods that are applicable to the RBS design and the RBS fuel.

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

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

Response: No.

The ECCS performance analysis methods are used to establish the APLHGR limits required by Technical Specification 3.2.1. The APLHGR limits are specified in the COLR and are the result of fuel design, design basis accident (DBA), and transient analyses. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis LOCA does not exceed the 2200 °F limit specified in 10 CFR 50.46.

The EXEM BWR–2000 evaluation model is an updated LOCA analytical method that has been approved by the NRC and is applicable to the RBS plant design and the fuel being used at RBS. A RBS plant specific ECCS performance analysis has been performed with the EXEM BWR–2000 evaluation model. This evaluation concluded that the resulting PCT still afforded adequate margin to the 2200 °F limit of 10 CFR 50.46.

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: Mark Wetterhahn, Esq., Winston & Strawn LLP, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: David Terao.

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

Date of amendment request: September 25, 2006.

Description of amendment request: The amendment proposes revisions to the Technical Specifications that are editorial in nature and consist of typographical corrections, update of references, and deletion of obsolete notes.

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 are editorial in nature and have no affect on accident scenarios previously evaluated. The proposed changes include typographical corrections, consistent with the current version of the Standard Technical Specifications (NUREG 1431, Revision 3); updated references, consistent with the current version of the Entergy Quality Assurance Program Manual (Revision 13); and deletion of notes that provided one-time allowances or are otherwise now obsolete. The proposed changes do not affect initiating events for accidents previously evaluated and do not affect or modify plants systems or procedures used to mitigate the progression or outcome of those accident scenarios.

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 do not involve the installation of new plant equipment or modification of existing plant equipment. No system or component setpoints are being changed and there are no changes being proposed for the way that the plant is operated. There are no new accident initiators or equipment failure modes resulting from the proposed changes. The proposed changes are editorial in nature, consisting of typographical corrections, reference updates, and deletion of obsolete notes.

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 are editorial in nature and do not change setpoints or limiting parameters specified in the plant Technical Specifications.

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. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Richard J. Laufer.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: August 31, 2006.

Description of amendment request: Entergy Operations, Inc., proposes to relocate Technical Specification (TS) 3.8.7 requirements associated with 120 Volt Inverter Y–28 and TS 3.8.9 requirements associated with 120 VAC electrical power distribution subsystem panel C–540 to the Technical Requirements Manual (TRM).

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 does not physically alter any plant structures, systems, or components and does not affect or create new accident initiators or precursors. The loss of Y–28, in itself, has no significant impact on station operation because its associated instrument panel, C–540, remains energized from an Emergency Diesel Generator (EDG) backed vital AC source. A potential loss of vital instrument panel C–540 does not prevent the fulfillment of a safety function and does not cause Emergency Safeguard Features (ESF) systems actuations that could render multiple ESF-related trains incapable of performing their intended safety function. Therefore, there is no effect on probability of accidents previously evaluated.

The proposed change relocates operability requirements for Y–28 and C–540 to the TRM. The TRM is part of the Safety Analysis Report (SAR) and is controlled under 10 CFR 50.59. In addition, TS-related components powered by C–540 continue to be governed by other TSs that limit the time in which the components can be out of service or provide compensatory measures during the out-of-service period.

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

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

Response: No.

The proposed change does not physically alter any structures, systems, or components, and does not affect or create new accident initiators or precursors. The accident analysis assumptions and results are unchanged. No new failures or interactions have been created. In addition, the proposed change does not introduce new failure modes or mechanisms associated with plant operation and will not create a new accident type.

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

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

Response: No.

The applicable margin of safety is the period of time that equipment important to safety is inoperable. There is no increase in risk that is a result of the proposed change because (1) affected non-TS components are not safety significant, (2) compensatory measures are procedurally established for those components governed by other regulation (i.e., 10 CFR [Part] 50, Appendix R), and (3) TS-related component out-of-service time or related compensatory actions are governed by other existing TSs. The proposed change does not affect any safety limits, other operational parameters, or setpoints in the TS, nor does it affect any margins assumed in the accident analyses. In addition, Y–28 and C–540 operability requirements will be relocated to the TRM, which is part of the Safety Analysis Report (SAR) and controlled by 10 CFR 50.59.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendments request: June 30, 2006.

Description of amendments request: The amendments would relocate the movable incore detectors and radioactive gaseous effluent oxygen monitoring instrumentation from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR).

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

Response: No.

The proposed amendment would relocate Technical Specification (TS) 3/4.3.3.2, ''Movable Incore Detectors,'' and TS 3/4.3.3.9 from the TS to the UFSAR. Movable Incore Detectors and Radioactive Gaseous Effluent Oxygen Monitoring Instrumentation are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. Movable Incore Detectors and Radioactive Gaseous Effluent Oxygen Monitoring Instrumentation are not accident mitigating structures, systems, or components. No impact on the plant response to accidents will be created. Thus the consequences of accidents previously analyzed are unchanged between the existing TS requirements and the proposed changes.

The proposed revision to TS SR [Surveillance Requirement] 4.11.2.5 is an administrative change to a reference necessitated by the proposed relocation of TS Table 3.3–13 from the TS to the UFSAR. The proposed revision to the TS Index, page renumbering, and minor format changes to improve consistency are also administrative changes necessitated by the proposed relocation of TS 3/4.3.3.2 and TS 3/4.3.3.9 from the TS to the UFSAR.

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

2. Do the proposed change[s] create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Specifically, no new hardware is being added to the plant as part of the proposed changes, no existing equipment is being modified, and no significant changes in operations are being introduced.

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

3. Do the proposed change[s] involve a significant reduction in a margin of safety?

Response: No.

The proposed changes will not alter any assumptions, initial conditions, or results of any accident analyses. The Movable Incore Detectors and oxygen monitoring instrumentation will continue to perform as before. The proposed changes relocate TS 3/4.3.3.2 and TS 3/4.3.3.9 from the TS to the UFSAR consistent with the guidance in NRC Generic Letter 95–10 and 10 CFR 50.36, and make conforming administrative changes to the TS Index, page renumbering, and minor format changes to improve consistency and to TS SR 4.11.2.5 to reflect the relocation of TS 3/4.3.3.9 from the TS to the Salem UFSAR.

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.

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

Date of amendments request: September 26, 2006.

Description of amendments request: The amendments would revise Technical Specification 6.9.1.9 to remove the revision number and date for the topical reports that contain the analytical methods used in the Core Operating Limits Report (COLR). The effect of this change is to allow the licensee to use current topical reports, as long as they have been approved by the NRC. The amendments would also add an NCR-approved topical report to the Salem Nuclear Generating Station, Unit No. 2, COLR methods.

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

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

Response: No.

The proposed changes affect the administrative controls section of Technical Specifications (TS) that govern the analytical methods used to determine core operating limits. Removal of revision levels and dates from NRC-approved methods listed in TS is an administrative change that has no impact on the probability or consequences of an accident. TS 6.9.1.9.b will still require these methods to be reviewed and approved by [the] NRC. The proposed change does not affect the required TS actions to be taken in the event that any core operating limits are exceeded.

The proposed use of WCAP–10054–P–A, Addendum 2 for the Salem Unit 2 Small Break Loss of Coolant Accident (SBLOCA) analysis is consistent with the limitations and conditions of NRC approval. The parameters assumed in the analysis are within the design limits of the plant equipment. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased, since it is shown that the Emergency Core Cooling System (ECCS) is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. No other accident is potentially affected by this change.

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

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

Response: No.

No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of the plant equipment. TS will continue to require operation within the core operating limits determined using NRC-approved analytical methods and the proposed change does not affect any actions required in the event the core operating limits are exceeded.

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

3. Do the proposed change[s] involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not have any impact on plant equipment or safety analysis acceptance criteria. Core operating limits will continue to be determined using NRC-approved analytical methods. The ECCS acceptance criteria of 10 CFR 50.46 will continue to be met following the proposed use of WCAP–10054–P–A, Addendum 2 for the Salem Unit 2 SBLOCA analysis[.]

Therefore, the proposed change[s] do[es] 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.

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

Date of amendment request: April 6, 2006.

Description of amendment request: The amendment would change the Technical Specifications to reduce the maximum allowable reactor power when two main steam safety valves (MSSVs) are inoperable.

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[es] the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change reduces the power level at which Salem Unit 2 may be operated with a maximum of two inoperable MSSVs in any steam generator. This change is consistent with analyses of the limiting transients for secondary system pressure (loss of load/turbine trip and rod withdrawal at power), performed to demonstrate the acceptance criterion of 110% of design pressure will continue to be met following steam generator replacement. The proposed change does not involve any changes to the MSSV actuation setpoints; they remain well above the Main Steam System operating pressures. The proposed change does not challenge the relief capacity of the MSSVs. Therefore, the probability of an event associated with mis-operation of the MSSVs (e.g., inadvertent depressurization of the Main Steam System) is not impacted by the proposed change. The proposed reduction in allowable power level establishes initial conditions consistent with the safety analyses, and does not affect the probability of any previously evaluated accident.

The proposed change is necessitated by analyses of limiting secondary system pressure transients, whose acceptance criteria continue to be met provided that plant operation is restricted to 58% RTP [rated thermal power] with a maximum of two inoperable MSSVs in any steam generator. There is no impact on any radiological consequences of an accident associated with the proposed reduction in maximum power level.

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

2. Do[es] the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Reducing the allowable power level per the proposed change does not introduce any new accident scenarios or malfunctions. The proposed change establishes an operating restriction consistent with current safety analysis methodology. It represents a change to an input assumption used in analyses of limiting secondary pressurization transients to ensure plant operation does not challenge any design limits.

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

3. Do[es] the proposed change involve a significant reduction in a margin of safety?

Response: No.

Acceptable margins of safety are inherent in the safety analysis acceptance criteria, including the limit on secondary system pressure to 110% of design pressure during a loss of load/turbine trip (LOL/TT) or rod withdrawal at power (RWAP) transient. The purpose of the proposed change is to limit operation with a maximum of two inoperable MSSVs for any steam generator, such that the acceptance criterion for secondary pressure continues to be met. The proposed change does not modify any acceptance criteria, nor would it cause any design limit to be exceeded.

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

NRC Branch Chief: Harold K. Chernoff.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: September 29, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification 3.7.8, ''Service Water (SW) System,'' to change the limiting conditions for operation (LCOs), Actions, Completion Times, and Surveillance Requirements (SRs). Specifically, the proposed amendment would change the LCO to require a specific number of SW pumps to be operable rather than the current SW train operability. The LCO Actions, Completion Times, and SRs would also be revised based on pump operability.

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 safety related function of the Service Water (SW) System is to provide cooling for safety related equipment, mitigate the containment response effects of a Main Steam Line Break (MSLB) and design basis Loss of Coolant Accident (LOCA), and provide long term containment and core cooling in the event of a LOCA. The operation of the SW system, including the number of pumps operating or available, has no affect on the probability of these accidents.

The probability of a loss of SW event is not increased. The proposed TS provides for more restrictive actions for pump inoperability than the existing TS, thereby reducing the probability of this event.

The consequences of a[n] MSLB or LOCA or other design basis accidents are not increased beyond that assumed in the accident analysis. Two service water pumps are sufficient for all accident mitigation functions. The change provides for adequate service water supply (2 pumps) for both normal and accident conditions. The availability of associated power supplies is also considered. For a reduction in the total number of available pumps, appropriate LCO action statements ensure that the pumps are returned to service within a time limit commensurate with an acceptable level of plant safety and risk, or the plant is placed in a safe mode.

The loss of SW has been previously evaluated and measures implemented to mitigate the event. Since a loss of SW assumes no SW pumps are operating, the proposed amendment has no affect on consequences of this event.

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

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

Response: No.

The only accidents directly initiated from this system are the loss of SW or flooding concerns. Both of these accidents have been previously evaluated with acceptable results. Therefore, this change does not create the possibility of a new or different [kind] of accident from any accident previously evaluated.

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

Response: No.

This change will ensure that sufficient SW pumps are available for accident mitigation at any one time while still providing the appropriate operational flexibility. A risk determination demonstrates that any increase in risk associated with this change is within the established regulatory guidelines. The technical analysis shows that appropriate action statements exist to ensure adequate SW is available for accident mitigation, considering emergency power supply availability. Therefore, this proposed change does not involve a significant reduction in [a] margin of safety.

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

Attorney for licensee: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Branch Chief: Richard J. Laufer.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: October 12, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.3.3, ''Capacity,'' to change the limit on the number of fuel assemblies in the spent fuel pool. The proposed amendment would also revise TS 3.7.13, ''Spent Fuel Pool Storage,'' to remove the references to Type 4 spent fuel pool storage racks, which are not currently installed.

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 reduces the total number of fuel assemblies that can be stored in the current spent fuel pool storage locations and reduces the number of available locations. This will limit the potential inventory of spent fuel in the pool. The probability of an accident has not changed since the number of stored fuel assemblies is not a precursor for a spent fuel handling accident. A comparison of the criticality analysis of fuel assemblies to be used in subsequent Extended Power Uprate core reloads to the current criticality analysis has been performed. The design parameter assumptions used in the licensing basis criticality analyses are bounding.

There are no new components or new functions associated with the spent fuel cooling system so the probability of an accident has not changed. The effect of a single failure on the spent fuel pool system's capability to provide for heat removal from the fuel pool has been analyzed. The analysis concluded that the system remains within the parameters previously evaluated. The implementation of the Extended Power Uprate does not affect the capability of the system to perform its function.

The Extended Power Uprate conditions do not add any new or previously unevaluated materials to the spent fuel pool storage system and do not include any reductions in the boron concentration requirements so the probability of an accident has not changed. The total soluble boron concentration required to maintain the spent fuel pool in a subcritical condition with the transition to the new fuel has not changed. The conclusions in the Ginna UFSAR [Updated Final Safety Analysis Report], assuming the most limiting accident, remain valid.

Therefore, the consequences of a fuel handling accident, a loss of spent fuel cooling, and a boron reduction concentration event previously evaluated have not increased.

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

Response: No.

The proposed changes do not alter the function of the spent fuel pool or any related equipment, nor cause it to operate differently than it was designed to operate. All equipment required to mitigate the consequences of an accident would continue to operate as before. The proposed changes reduce the maximum number of fuel assemblies that can be stored in the spent fuel pool and the number of storage locations. Therefore, this change does not create the possibility of a new or different [kind] of accident from any accident previously evaluated.

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

Response: No.

The proposed changes reduce the maximum number of fuel assemblies that can be stored in the spent fuel pool and the number of storage locations. The changes are in accordance with conclusions supporting Extended Power Uprate and have been determined to be acceptable. The design parameter assumptions used in the licensing basis criticality analysis bound those of the new fuel assemblies. Although the individual heat load per assembly has increased due to the changed fuel design, the maximum spent fuel pool heat load has decreased due to the reduction in the number of fuel assemblies that will be stored based on future plans to use dry cask storage. Therefore, this proposed change does not reduce 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: October 3, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specifications surveillance requirements related to inspection of the containment sump trash racks and screens, inside recirculation spray (RS) pump wells, and outside RS and low head safety injection pump suction inlets resulting from Nuclear Regulatory Commission's (NRC's) Generic Safety Issue (GSI) 191 and Generic Letter (GL) 2004–02.

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?

The proposed change does not impact the condition or performance of any plant structure, system or component. Furthermore, the proposed change does not affect the initiators of any previously analyzed event or the assumed mitigation of accident or transient events since the plant will be operated in the same manner and within the same operating limits that are currently in place. The proposed TS change is administrative in nature given that inspection of containment sump components for debris accumulation and structural integrity will continue to be performed. The revised TS surveillance wording is being implemented as a clarification to facilitate inspection of the containment sump in its current configuration, as well as after containment sump modifications have been implemented in response to GSI–191 and GL 2004–002. As a result, the proposed change to the Surry TS does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by this proposed change.

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

The proposed change is administrative in nature and, as such, does not involve any changes in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints, that initiate protective or mitigative actions, since the plant will be operated in the same manner and within the same operating limits that are currently in place. Since plant operation will not be affected by this change, no new failure modes are being introduced. Therefore, the proposed change to the Surry TS does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

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

The proposed change is administrative in nature given that inspection of the containment sump components for debris accumulation and structural integrity will continue to be performed on an established frequency. The more general nature of the TS surveillance wording is being implemented as a clarification to facilitate inspection of the containment sump in its current configuration, as well as after containment sump modifications have been implemented in response to GSI–191 and GL 2004–002. The proposed change does not impact station operation or any plant structure, system or component that is relied upon for accident mitigation. Furthermore, the margin of safety assumed in the plant safety analysis is not affected in any way by the proposed change since the plant will be operated in the same manner and within the same operating limits and setpoints that are currently in place. Therefore, the proposed change to the Surry Technical Specifications does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Branch Chief: Evangelos C. Marinos.

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 51.12(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 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.*

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

Date of application for amendment: November 30, 2005.

Brief description of amendment: The amendment revises the surveillance requirements (SR) for the emergency diesel generator automatic trips bypass of SR 3.8.1.11 from 18 months to 24 months.

Date of issuance: October, 4, 2006.

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

Amendment No. 208.

Renewed Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 28, 2006 (71 FR 10072).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 2006.

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

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

Date of application for amendments: July 27, 2005, as supplemented May 4, 2006, and August 8, 2006.

Brief description of amendments: The amendments revise the Catawba and McGuire Technical Specification 3.4.15, ''RCS Leakage Detection Instrumentation.''

Date of issuance: September 30, 2006.

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

Amendment Nos.: 234/230 and 235/217.

Renewed Facility Operating License Nos. NPF–35, NPF–52, NPF–9 and NPF–17: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register:*** August 30, 2006 (71 FR 51644).

The supplement dated August 8, 2006, provided clarifying information that did not expand the scope of the July 27, 2005, application as modified May 4, 2006.

The Commission's related evaluation, Final No Significant Hazards Finding, and State consultation of the amendments are contained in a Safety Evaluation dated September 30, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. STN 50–457, Braidwood Station, Unit No. 2, Will County, Illinois

Date of application for amendment: November 18, 2005, as supplemented by letters dated August 18 and September 28, 2006.

Brief description of amendment: The amendment revised TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' regarding the required SG inspection scope for Braidwood Station, Unit No. 2, during refueling outage 12 and the subsequent operating cycle.

Date of issuance: October 24, 2006.

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

Amendment No.: 141.

Facility Operating License No. NPF–77: The amendment revised the Technical Specifications and License.

Date of initial notice in ***Federal Register:*** (71 FR 29676; May 23, 2006).

The August 18 and September 28, 2006, supplements contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 2006.

No significant hazards consideration comments received: No.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: March 23, 2006.

Description of amendment request: The amendment deleted License Condition 2.G, ''Reporting to the Commission,'' as described in the Notice of Availability published in the **Federal Register** on April 25, 2006 (71 FR 23955). The change was requested as part of the consolidated line item improvement process and consistent with the model safety evaluation published in the **Federal Register** on November 4, 2005 (70 FR 67202).

Date of issuance: October 17, 2006.

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

Amendment No.: 113.

Facility Operating License No. NPF–86: The amendment revised Facility Operating License No. NPF–86 and the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 25, 2006 (71 FR 23955).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 17, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 16, 2006.

Brief description of amendment: The amendment revised the Technical Specification 3.10.1, ''Inservice Leak and Hydrostatic Testing Operation,'' to extend the scope to include provisions for temperature increases above 212 °F as a consequence of inservice leak or hydrostatic testing, and as a consequence of control rod scram time testing initiated in conjunction with the inservice leak test or hydrostatic test, when initial test conditions are below 212 °F.

Date of issuance: October 23, 2006.

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

Amendment No.: 225.

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

Date of initial notice in ***Federal Register:*** August 1, 2006 (71 FR 43535)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 7, 2005, as supplemented by letters dated July 20 and September 5, 2006.

Brief description of amendments: These amendments revised the Technical Specifications to delete Surveillance Requirement (SR) 4.9.2.b, which requires performance of a channel functional test (CFT) of each source range neutron flux monitor within 8 hours prior to the initial start of core alterations. An associated administrative change would renumber current SR 4.9.2.c as SR 4.9.2.b. The amendments would also eliminate the restriction in SRs 4.10.3.2 and 4.10.4.2 that the CFTs of the intermediate and power range monitors be performed within 12 hours prior to initiating physics tests.

Date of issuance: October 13, 2006.

Effective date: As of the date of issuance, to be implemented in 60 days.

Amendment Nos.: 275, 257.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications and License.

Date of initial notice in ***Federal Register:*** August 2, 2006 (71 FR 43819). The supplements 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 October 13, 2006.

No significant hazards consideration comments received: No.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: November 15, 2005, as supplemented May 31, August 31, and September 29, 2006.

Brief description of amendment: The amendment revises the Virgil C. Summer Nuclear Station Technical Specifications (TS) 3/4.3 for the reactor trip instrumentation and the engineered safety feature actuation system instrumentation to implement the allowed outage time and bypass test time changes approved in WCAP–14333–P–A, Revision 1, ''Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,'' and makes several additional changes to TS outside of the scope of WCAP–14333.

Date of issuance: October 24, 2006.

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

Amendment No. 177.

Renewed Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 20, 2005 (70 FR 75496).

The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 2006.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–259 Browns Ferry Nuclear Plant, Unit 1, Limestone County, Alabama

Date of application for amendment: November 10, 2003 (TS–430), as supplemented by letter dated November 8, 2004.

Brief description of amendment: The amendment incorporates the necessary Technical Specification (TS) changes for the planned replacement of the power range monitoring portion of the existing Neutron Monitoring System with a digital upgrade. These changes expand the current allowable operating domain to the Maximum Extended Load Line Limit region of the power/flow chart.

Date of issuance: September 27, 2006.

Effective date: Date of issuance, to be implemented within 30 days.

Amendment No.: 262.

Facility Operating License No. DPR–33: Amendment revised the TSs.

Date of initial notice in ***Federal Register:*** February 3, 2004 (69 FR 5208). The November 8, 2004, supplement, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 27, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 6, 2006.

Brief description of amendments: The amendments modify Technical Specification (TS) requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.7. This operating license improvement was made available by the Nuclear Regulatory Commission (NRC) on May 4, 2005 (70 FR 23252) as part of the consolidated line item improvement process and is consistent with NRC approved Technical Specification Task Force (TSTF) standard TS change TSTF–372, Revision 4.

Date of issuance: October 4, 2006.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos. 312/301.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the technical specifications.

Date of initial notice in ***Federal Register:*** March 28, 2006 (71 FR 15487).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 30th day of October 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–18595 Filed 11–6–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Model License Amendment Request and Safety Evaluation on Technical Specification Improvement Regarding Revision to the Completion Time in STS 3.6.6A, ''Containment Spray and Cooling Systems'' for Combustion Engineering Pressurized Water Reactors Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: Notice is hereby given that the staff of the U.S. Nuclear Regulatory Commission (NRC) has prepared a model license amendment request (LAR), model safety evaluation (SE), and model proposed no significant hazards consideration (NSHC) determination related to changes to the completion times (CT) in Standard Technical Specification (STS) 3.6.6A, ''Containment Spray and Cooling Systems,'' contained in NUREG–1432 (Standard Technical Specifications for Combustion Engineering Plants, Rev. 3.0). The proposed changes would revise STS 3.6.6A by extending the CT for one containment spray system (CSS) train inoperable from 72 hours to seven days, and add a Condition, Required Actions and associated CT when one CSS train and one containment cooling system (CCS) train are inoperable. These changes are based on analyses provided in a joint applications report submitted by the Combustion Engineering Owner's Group (CEOG). The CEOG participants in the Technical Specifications Task Force (TSTF) proposed these changes to the STS in Change Traveler No. TSTF–409, Revision 2.

The purpose of these models is to permit the NRC to efficiently process amendments to incorporate these changes into plant-specific STS for Combustion Engineering pressurized water reactors (PWRs). Since TSTF–409 involves a risk-informed approach to extending the CT for one CSS inoperable, the NRC staff must verify that licensees who apply for this TS change have a valid, up-to-date probabilistic risk assessment (PRA) model that employs PRA principles to ensure that public health and safety are maintained when the CSS CT of 7 days is implemented. Therefore, the model LAR contains several conditions requiring licensees to make specific validations of their plant PRA quality and methods. The intent of using the CLIIP to adopt TSTF–409 is to eliminate