advice to the Commission with regard to the hazards of proposed or existing reactor facilities, to review each application for a construction permit or operating license for certain facilities specified in the AEA, and such other duties as the Commission may request. The AEA as amended by Public Law 100–456 also specifies that the Defense Nuclear Safety Board may obtain the advice and recommendations of the ACRS.

Membership on the Committee includes individuals experienced in reactor operations, management; probabilistic risk assessment; analysis of reactor accident phenomena; design of nuclear power plant structures, systems and components; materials science; and mechanical, civil, and electrical engineering.

The Nuclear Regulatory Commission has determined that renewal of the charter for the ACRS until December 12, 2008 is in the public interest in connection with the statutory responsibilities assigned to the ACRS. This action is being taken in accordance with the Federal Advisory Committee Act.

FOR FURTHER INFORMATION CONTACT: Andrew L. Bates, Office of the Secretary, NRC, Washington, DC 20555; telephone: (301) 415–1963.

Dated: December 13, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–21583 Filed 12–18–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of December 18, 25, 2006, January 1, 8, 15, 22, 2007.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of December 18, 2006

Thursday, December 21, 2006

12:55 p.m. Affirmation Session (Public Meeting) (Tentative) a. Entergy Nuclear Vermont Yankee, LLC, & Entergy Nuclear Operations, Inc. (Vermont Yankee Nuclear Power Station), LBP–06–20 (Sept. 22, 2006), reconsid'n denied (Oct. 30, 2006) (Tentative). b. Final Rulemaking to Revise 10 CFR 73.1, Design Basis Threat (DBT) Requirements (Tentative).

Week of December 25, 2006—Tentative

There are no meetings scheduled for the Week of December 25, 2006.

Week of January 1, 2007—Tentative

Thursday, January 4, 2007

12:55 p.m. Affirmation Session (Public Meeting) (Tentative) a. Entergy Nuclear Operations, Inc. (Pilgrim Nuclear Power Station), Intervenor Pilgrim Watch's Appeal of LBP–06–23 (Ruling on Standing and Contentions) (Tentative).

Week of January 8, 2007—Tentative

Wednesday, January 10, 2007

9:30 a.m. Briefing on Browns Ferry Unit 1 Restart (Public Meeting) (Contact: Catherine Haney, 301–415–1453).

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

Thursday, January 11, 2007

1:30 p.m. Periodic Briefing on New Reactor Issues (Public Meeting) (Contact: Donna Williams, 301–415–1322).

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

Week of January 15, 2007—Tentative

There are no meetings scheduled for the Week of January 15, 2007.

Week of January 22, 2007—Tentative

Tuesday, January 23, 2007

1:30 p.m. Joint Meeting with Federal Energy Regulatory Commission on Grid Reliability (Public Meeting) (Contact: Mike Mayfield, 301–415–5621).

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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Additional Information

Affirmation of Entergy Nuclear Vermont Yankee, LLC, & Entergy Nuclear Operations, Inc. (Vermont Yankee Nuclear Power Station), LBP–06–20 (Sept. 22, 2006), reconsid'n denied (Oct. 30, 2006) (Tentative) tentatively scheduled on Thursday, December 14, 2006 at 9:25 a.m. has been rescheduled tentatively on Thursday, December 21, 2006 at 12:55 p.m.

* * * * *

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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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you 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, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by e-mail at *DLC@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, D.C. 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: December 13, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–9787 Filed 12–15–06; 1:56 pm]

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 22, 2006 to December 7, 2006. The last biweekly notice was published on December 5, 2006 (71 FR 70553).

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*.

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

Date of amendment request: September 15, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 6.8.5, ''Reactor Building Leakage Rate Testing Program,'' to allow a one-time deferral of the next Type A, containment integrated leak rate test (ILRT) from ''no later than September 2008'' to ''prior to startup from T1R18 refueling outage.'' The NRC has previously approved a one-time 5-year extension to the Type A ILRT schedule for TMI–1 by issuance of Amendment No. 244, dated August 14, 2003. Amendment No. 244 changed the TSs to state that the Type A ILRT shall be performed no later than September 2008. The proposed amendment would add approximately 15 months to the currently-approved 15-year interval. This deferral would allow the Type A ILRT to be performed during a steam generator replacement in the fall of 2009.

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:

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

Response: No.

The proposed change will revise TS 6.8.5 to reflect a one-time extension to the Three Mile Island, Unit 1 Type A Integrated Leak Rate Test (ILRT) as currently specified in the Technical Specifications. This change will extend the requirement to perform the Type A ILRT from the current requirement of ''no later than September 2008'' to ''prior to startup from the T1R18 refueling outage,'' which is currently scheduled for Fall 2009. The current Type A ILRT interval of 15 years, based on past performance, would be extended on a one-time basis by approximately 15 months.

The function of the containment is to isolate and contain fission products released from the reactor coolant system following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A ILRTs is not a precursor of any accident previously evaluated. Type A ILRTs provide assurance that the TMI, Unit 1 containment will not exceed allowable leakage rate values specified in the TS and will continue to perform its design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in postulated total population dose rate and an insignificant increase in the postulated conditional containment failure probability. Additionally, containment inspections have also been performed which demonstrate the continued structural integrity of the primary containment.

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

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 for a one-time extension of the Type A ILRT for TMI, Unit 1 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

Response: No.

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by a Type A ILRT, as required by 10 CFR [Part] 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' These tests are performed to verify the essentially leak tight characteristics of the containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A ILRT does not affect the method for Type A, B or C testing or the test acceptance criteria.

AmerGen has conducted a risk assessment to determine the impact of a change to the TMI, Unit 1 Type A ILRT schedule from a baseline ILRT frequency of three times in 10 years to once in 15 years plus 15 months for the risk measures of Large Early Release Frequency (*i.e.*, LERF), Population Dose, and Conditional Containment Failure Probability (*i.e.*, CCFP). This assessment indicated that the proposed TMI, Unit 1 ILRT interval extension has a small change in risk to the public and is an acceptable plant change from a risk perspective.

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

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

Attorney for licensee: Mr. Brad Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: May 30, 2006, as supplemented by letter dated November 20, 2006.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) requirements related to steam generator tube integrity. The amendment would adopt Nuclear Regulatory Commission (NRC)-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

The NRC staff published a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated May 30, 2006.

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 change requires a SG [Steam Generator] Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A Steam Generator Tube Rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as Main Steam Line Break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TSs identifies the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TSs. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design-basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than 500 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed change does not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TSs.

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 (Acting): Douglas V. Pickett.

Carolina Power & Light Company, Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: May 23, 2006, as supplemented by letter dated October 3, 2006.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) requirements related to steam generator tube integrity. The amendment would adopt Nuclear Regulatory Commission (NRC)-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

The NRC staff published a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated May 23, 2006.

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 change requires a SG [Steam Generator] Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A Steam Generator Tube Rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as Main Steam Line Break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TSs identifies the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TSs. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design-basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than 500 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed change does not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TSs.

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 (Acting): Douglas V. Pickett.

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

Date of amendment request: June 1, 2006, as supplemented by letter dated November 20, 2006.

Description of amendment request: The proposed amendment would revise the surveillance requirements (SR) for the emergency core cooling system suction inlet in the containment as specified in Technical Specification SR 3.5.2.6.

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?

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed surveillance change will continue to ensure that the emergency core cooling system (ECCS) containment sump inlet is inspected in a manner that will verify operability. Performance of the required system surveillances, in conjunction with the applicable operational and design requirements for the ECCS, provide assurance that the system will be capable of performing the required design functions for accident mitigation and that the system will perform in accordance with the functional requirements for the system as described in the Updated Final Safety Analysis Report for HBRSEP, Unit No. 2. The proposed rewording of the surveillance requirement will continue to ensure that the ECCS containment sump suction inlet is not restricted by debris and suction inlet strainers show no evidence of structural distress or abnormal corrosion for HBRSEP, Unit No. 2. This ensures that the rate of occurrence and consequences of analyzed accidents will not 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 previously evaluated?

No. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. HBRSEP, Unit No. 2, is replacing the existing ECCS containment sump inlet trash racks and screens with new strainers in accordance with the response to Generic Letter 2004–02. The strainer is a passive component in the ECCS, which is a standby safety system used for accident mitigation. As such, the strainer cannot be an accident initiator. A change to Technical Specifications Surveillance Requirement 3.5.2.6 is needed to accommodate the change to the ECCS containment sump inlet design. This change does not alter the nature of events postulated in the HBRSEP, Unit No. 2, Updated Final Safety Analysis Report, nor does it introduce any unique precursor mechanisms. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change does not involve a significant reduction in the margin of safety. The proposed change to the ECCS containment sump inlet surveillance requirement provides appropriate and applicable surveillance for this system. The proposed change to this surveillance requirement for the ECCS system will continue to ensure system operability. The proposed change does not adversely affect any plant safety limits, setpoints, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System (RCS), or containment integrity. Therefore, this change does not affect any margin of safety for HBRSEP, Unit No. 2.

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 (Acting): Douglas Pickett.

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

Date of amendment request: May 31, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator (SG) tube integrity. In particular, Dominion Nuclear Connecticut, Inc. (DNC) is proposing to replace the existing SG tube surveillance program with the NRC-approved Technical Specifications Task Force (TSTF) 449, Revision 4. The proposed changes are consistent with the Consolidated Line Item Improvement Process (CLIIP) provided in the May 6, 2005, **Federal Register** notice (70 FR 24126). In addition, the Millstone Power Station, Unit No. 2 (MPS2) TSs are revised beyond the scope of the CLIIP to provide consistent terminology and format. *Basis for proposed no significant hazards consideration determination:* DNC proposed minor variations and/or deviations from the TS changes described in the CLIIP beyond the scope of the no significant hazards consideration determination published on March 2, 2005. DNC has evaluated the proposed beyond-scope TS changes and determined it does not represent a significant hazards consideration. As required by 10 CFR 50.91(a), DNC 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 changes do not affect initiators of previously analyzed events or assumed mitigation of accident or transient events.

Therefore, these changes do not involve a significant increase in 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.

The proposed changes involve adding a new definition and rewording the existing TS to be consistent with NUREG–1432, Revision 3. In addition, the requested change for MPS2 incorporates a more conservative leakage limit of 75 gallons per day per steam generator as opposed to the CLIIP specified limit of 150 gallons per day per steam generator. The changes will not impose any requirements or eliminate any existing requirements that will 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. Since the proposed changes do not have an impact on any safety analysis assumptions and accidents previously evaluated, there are no margin of safety issues involved.

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

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

Attorney for licensee: 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., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: May 31, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator (SG) tube integrity. In particular, Dominion Nuclear Connecticut, Inc. (DNC) is proposing to replace the existing SG tube surveillance program with the NRC-approved Technical Specifications Task Force (TSTF) 449, Revision 4. The proposed changes are consistent with the Consolidated Line Item Improvement Process (CLIIP) provided in the May 6, 2005, **Federal Register** notice (70 FR 24126). In addition, the Millstone Power Station, Unit No. 3 (MPS3) TSs are revised beyond the scope of the CLIIP to provide consistent terminology and format.

Basis for proposed no significant hazards consideration determination: DNC proposed minor variations and/or deviations from the TS changes described in the CLIIP beyond the scope of the no significant hazards consideration determination published on March 2, 2005. DNC has evaluated the proposed beyond-scope TS changes and determined it does not represent a significant hazards consideration. As required by 10 CFR 50.91(a), DNC has provided its analysis of the issue of no significant hazards consideration to support this conclusion. 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 changes involve rewording the existing technical specifications to be consistent with NUREG–1431, Revision 3. These proposed changes do not affect initiators of previously analyzed events or assumed mitigation of accident or transient events.

Therefore, these changes do not involve a significant increase in 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.

These proposed changes do not involve physical alteration of the plant (no new or different type of equipment will be installed). The changes will not impose any requirements or eliminate any existing requirements that will 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.

Since the proposed changes do not have an impact on any safety analysis assumptions and accidents previously evaluated, there are no margin of safety issues involved.

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

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

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

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: November 13, 2006.

Description of amendment request: The proposed amendment would revise Grand Gulf Nuclear Station, Unit 1, Technical Specification (TS) Limiting Condition for Operation (LCO) 3.10.1, and the associated TS Bases, to expand its scope to include provisions for temperature excursions greater than 200 °F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in MODE 4.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 21, 2006 (71 FR 48561), on possible amendments to revise the plant-specific TS, to expand the scope of TS LCO 3.10.1, to include provisions for temperature excursions greater than 200 °F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in MODE 4, including a model safety evaluation and model No Significant Hazards Determination (NSHC), using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 27, 2006 (71 FR 63050). The licensee affirmed the applicability of the model NSHC determination in its application dated November 13, 2006.

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

Technical Specifications currently allow for operation at greater than [200] °F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact the probability or consequences of an accident previously evaluated. 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

Technical Specifications currently allow for operation at greater than [200] °F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. No new operational conditions beyond those currently allowed by LCO 3.10.1 are introduced. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed 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

Technical Specifications currently allow for operation at greater than [200] °F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact any margin of safety. Allowing completion of inspections and testing and supporting completion of scram time testing initiated in conjunction with an inservice leak or hydrostatic test prior to power operation results in enhanced safe operations by eliminating unnecessary maneuvers to control reactor temperature and pressure. Therefore, the proposed 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: Terence A. Burke, Associate General Council—Nuclear, Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: David Terao.

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

Date of application for amendments: July 14, 2006.

Description of amendment request: The proposed changes would modify Technical Specification (TS) requirements related to required end states for TS action statements. The changes are generally consistent with the NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Change Traveler, TSTF–423, ''Risk Informed Modification to Selected Required Action End States for BWR Plants.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 14, 2005 (70 FR 74037), on possible amendments adopting TSTF–423, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 23, 2006 (71 FR 14726).

The licensee affirmed the applicability of the following TSTF–423 model NSHC determination in its application dated July 14, 2006.

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 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 [General Electric] NEDC–32988, Revision 2, ''Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR [boiling-water reactor] 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.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). 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.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in a 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 [Boiling Water Reactor Owner's Group's] risk assessment approach is comprehensive and follows staff guidance as documented in RGs [Regulatory Guides] 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 has reviewed the 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. Brad Fewell, Assistant General Counsel, Exelon Generation Company LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Harold K. Chernoff.

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

Date of application for amendments: September 15, 2006.

Description of amendment request: The proposed amendment would revise the required frequency for control rod scram time testing, as described in Technical Specification (TS) Surveillance Requirement 3.1.4.2, from ''120 days cumulative operation in MODE 1'' to ''200 days cumulative operation in MODE 1.'' The proposed TS change is based on the NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Change Traveler, TSTF–460, ''Control Rod Scram Time Testing Frequency.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on May 27, 2004 (69 FR 30339), on possible amendments adopting TSTF–460, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on August 23, 2004 (69 FR 51864).

The licensee affirmed the applicability of the following TSTF–460 model NSHC determination in its application dated September 15, 2006.

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

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The frequency of surveillance testing is not an initiator of any accident previously evaluated. The frequency of surveillance testing does not affect the ability to mitigate any accident previously evaluated, as the tested component is still required to be operable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change does not result in any new or different modes of plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change continues to test the control rod scram time to ensure the assumptions in the safety analysis are protected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the 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 NSHC.

Attorney for Licensee: Mr. Brad Fewell, Assistant General Counsel, Exelon Generation Company LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: November 14, 2006.

Description of amendment request: The proposed amendment would revise Specification 3.3.5.1–1 of the Technical Specifications (TSs) to permit a one-time extension of the quarterly surveillance interval (i.e., from 92 days to 140 days) for three low pressure coolant injection (LPCI) loop select logic functions.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) Part 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC), 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.

This amendment requests a one-time extension to the performance interval for a limited number of TS surveillance requirements. The performance of these surveillances, or the failure to perform, is not a precursor and does not affect the probability of an accident. Therefore, the delay in performance proposed in this amendment request for these surveillance requirements does not increase the probability of an accident previously evaluated.

A delay in performing these surveillances does not result in a system being unable to perform its required function. In the case of this one-time extension, the relatively short period of additional time period for the systems and components to be in service prior to the next performance of the surveillance will not affect the ability of those systems to operate as designed. Therefore, the systems required to mitigate accidents will remain capable of performing their required function. No new failure modes have been introduced because of this action and the consequences remain consistent with previously evaluated accidents. Therefore, the proposed delay in performance of the surveillance requirements in this amendment request does not involve a significant increase in the consequences of an accident.

Therefore, operation of the facility in accordance with the proposed license amendment would 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 or accident from any accident previously evaluated?

Response: No.

The proposed amendment does not involve a physical alteration of any system, structure, or component (SSC) or a change in the way any SSC is operated. The proposed amendment does not involve operation of any SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the one-time surveillance requirement deferrals being requested.

Thus, 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 proposed amendment is a one-time extension of the performance interval of a limited number of TS surveillance requirements. Extending these surveillance requirements does not involve a modification of any TS Limiting Condition for Operation. Extending these surveillance requirements does not involve a change to any limit on accident consequences specified in the license or regulations. Extending these surveillance requirements does not involve a change to how accidents are mitigated or a significant increase in the consequences of an accident. Extending these surveillance requirements does not involve a change in a methodology used to evaluate consequences of an accident. Extending these surveillance requirements does not involve a change in any operating procedure or process.

The instrumentation and components involved in this request have exhibited reliable operation based on the results of their performance during past periodic ECCS [emergency core cooling system] functional testing.

Based on the limited additional period of time that the systems and components will be in service before the surveillances are next performed, as well as the operating experience that these surveillances are typically successful when performed, it is reasonable to conclude that the margins of safety associated with these surveillance requirements will not be affected by the requested extension.

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

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

Attorney for licensee: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Branch Chief: L. Raghavan.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: November 6, 2006.

Description of amendment request: The proposed amendment would add the realistic large break loss-of-coolant accident (RLBLOCA) methodology to the analytical methods referenced in Technical Specification (TS) 5.6.5.b.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (CFR) Section 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 license amendment adds approved analytical methods used to determine the core operating limits per Technical Specification 5.6.5.b. Accidents previously evaluated will be unaffected because they will continue to be analyzed using applicable methodologies approved by the Nuclear Regulatory Commission to ensure all required safety limits are met. The proposed amendment does not affect the acceptance criteria for any Final Safety Analysis Report (FSAR) safety analysis analyzed accidents and anticipated operational occurrences. As such, the proposed amendment does not increase the probability or consequences of an accident. The proposed amendment does not involve operation of the required structures, systems or components (SSCs) in a manner or configuration different from those previously recognized or evaluated.

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

Response: No.

The proposed amendment does not involve a physical alteration of any SSC or a change in the way any SSC is operated. The proposed amendment does not involve operation of any required SSCs 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 amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment does not, by itself, introduce a failure mechanism. The proposed amendment does not involve any physical changes to the plant or manner in which the plant is operated. The proposed changes do not affect the acceptance criteria for any FSAR safety analysis analyzed accidents or anticipated operational occurrences. All required safety limits would continue to be analyzed using methodologies approved by the Nuclear Regulatory Commission.

Therefore, the proposed amendment would 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: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Branch Chief: L. Raghavan.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: November 13, 2006.

Description of amendment request: The proposed amendment would relocate the requirements of Technical Specification (TS) 2.22, ''Toxic Gas Monitors,'' and TS Table 3–3, Item 29 to the Fort Calhoun Station, Unit No. 1, Updated Safety Analysis Report (USAR).

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

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

Response: No.

The proposed change relocates requirements for toxic gas monitors that do not meet the criteria for inclusion in the TS set forth in 10 CFR 50.36(c)(2)(ii). The requirements for toxic gas monitors are being relocated from [the] TS to the USAR, which will be maintained pursuant to 10 CFR 50.59, thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Response: No.

The proposed change relocates requirements for toxic gas monitors that do not meet the criteria for inclusion in [the] TS set forth in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does th[e] [proposed] change involve a significant reduction in a margin of safety?

Response: No.

The proposed change relocates requirements for toxic gas monitors that do not meet the criteria for inclusion in [the] TS set forth in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements for toxic gas monitors remain the same as the existing TS. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety. [Therefore, the TS change does not involve a significant reduction in the margin of safety.]

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: May 31, 2006.

Description of amendment request: The proposed amendments would correct administrative errors in the SSES 1 and 2 Technical Specifications (TSs) by adding a logical connector in Condition B of Limiting Condition for Operation (LCO) 3.8.1 (SSES 1 TS only) and correct the routing of Interstate 80 (I–80) on Figure 4.1–2 in the SSES 1 and 2 TSs Section 4.0.

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.

Change to Technical Specification 3.8.1

The proposed change is administrative in nature and does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

Change to Technical Specification Figure 4.1–2

The proposed change is administrative in nature and does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment or any design changes to the facility.

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.

Change to Technical Specification 3.8.1

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

Change to Technical Specification Figure 4.1–2

The proposed change is an administrative change and will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. 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.

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

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

Response: No.

Change to Technical Specification 3.8.1

The proposed change revises Condition B in LCO 3.8.1 to be consistent with Technical Specification 1.2, ''Logical Connectors.'' This change is administrative in nature. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

Change to Technical Specification Figure 4.1–2

The proposed change is administrative in nature and does not affect any plant systems.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: September 7, 2006.

Description of amendment request: The proposed amendments would revise the SSES 1 and 2 Technical Specification (TSs) Section 5.5.6, ''Inservice Testing Program,'' and TS 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to be consistent with the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(f)(4) and 10 CFR 50.55a(g)(4), respectively. The proposed amendments would implement TS Task Force (TSTF) 343, Revision 1 and TSTF 479, 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.

Change to Technical Specification 5.5.6

The proposed change revises the Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3.

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

Change to Technical Specification 5.5.12

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR [Part] 50, paragraph 55a (g)(4) for components classified as Code Class CC.

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 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 pursuant to NRC approved ASME Section XI Code requirement (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous 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 does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment, or any design changes to the facility.

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.

Change to Technical Specification 5.5.6

The proposed change revises the Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3.

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

Change to Technical Specification 5.5.12

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR [Part] 50, paragraph 55a (g)(4) for components classified as Code Class CC.

The change affects the frequency of visual examinations that will be performed for the concrete surfaces containments. In addition, the proposed change 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 plant (*i.e.*, no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. 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.

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

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

Response: No.

Change to Technical Specification 5.5.6

The proposed change revises the Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The safety function of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

Change to Technical Specification 5.5.12

The proposed change revises the Improved Standard Technical Specification Administrative Controls program requirements for consistency with the requirements of 10 CFR [Part] 50, paragraph 55a (g)(4) for components classified as Code Class CC.

The change affects the frequency of visual examinations that will be performed for the concrete surfaces of containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The safety function of the containment as a fission product barrier will be maintained.

[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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2 (SSES 2), Luzerne County, Pennsylvania

Date of amendment request: November 16, 2006.

Description of amendment request: The proposed amendment would revise the SSES 2 Technical Specification (TS) Section 2.1.1.2 to reflect the Unit 2 Cycle 14 (U2C14) Minimum Critical Power Ratio (MCPR) Safety Limits for two-loop and single-loop operation. Additionally, TS Section 5.6.5.b would be revised to reflect the NRC-approved methodology used in the MCPR Safety Limit Analysis.

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:

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

Response: No.

The proposed change to the two-loop and single-loop MCPR Safety Limits do not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. Further, the proposed U2C14 MCPR Safety Limits were generated using NRC-approved methodology and meet the applicable acceptance criteria. Thus, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

Prior to the startup of U2C14, licensing analyses are performed (using NRC-approved methodology referenced in Technical Specification Section 5.6.5.b) to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are added to the MCPR Safety Limit values to generate the MCPR operating limits in the U2C14 COLR [Core Operating Limits Report]. These limits could be different from those specified for the previous Unit 2 COLR. The COLR operating limits thus assure that the MCPR Safety Limit will not be exceeded during normal operation or anticipated operational occurrences. Postulated accidents are also analyzed prior to the startup of U2C14 and the results shown to be within the NRC-approved criteria.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC-approved methodology used to generate the U2C14 core operating limits. The use of this approved methodology does not increase the probability [* * *] or consequences of an accident previously evaluated.

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

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

Response: No.

The changes to the two-loop and single-loop MCPR Safety Limits do not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC-approved methodology used to generate the U2C14 core operating limits. The use of this approved methodology does not create the possibility of a new or different kind of accident.

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

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

Response: No.

Since the proposed changes do not alter any plant system, equipment, component, or the processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed two-loop and single-loop MCPR Safety Limits do not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections because the MCPR Safety Limits calculated for U2C14 preserve the required margin of safety.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC-approved methodology used to generate the U2C14 core operating limits. This approved methodology is used to demonstrate that all applicable criteria are met, thus, demonstrating that there is no reduction in 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 amendment request involves no significant hazards consideration.

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

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 amendment request: August 4, 2006.

Description of amendment request: The amendments would revise the Technical Specifications (TSs) to allow the movement of irradiated fuel inside containment to commence at 24 hours after shutdown or at the decay time calculated using the licensee's spent fuel pool integrated decay heat management program, whichever is later.

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 of occurrence 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 in less time after subcriticality (but more accurately calculated), 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 proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated. The accident previously evaluated that is associated with the proposed license amendment is the fuel handling accident. Allowing the fuel to be offloaded based on the IDHM [integrated decay heat management program] calculated time after subcriticality 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 fuel handling accident. 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 Reg. Guide 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 [fuel handling accident] analysis is 24 hours.

Therefore, the proposed license amendment does not increase the probability of occurrence or the consequences of accidents previously evaluated are not increased.

2. Create 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 (but more accurately calculated), which 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 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 fuel handling accident. 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 demonstrated that the existing Spent Fuel Pool cooling system and associated systems under worst-case circumstances would maintain the integrity of the Spent Fuel Pool. The proposed method of offload does not create a new or different kind of accident from any accident previously evaluated.

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 fuel handling accident. The shorter decay time prior to fuel movement has been evaluated against 10 CFR part 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 heatup calculations performed prior to each refueling outage as part of the IDHM program ensure that planned spent fuel transfer to the SFP [spent fuel pool] 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 Spent Fuel Pool 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 Reg. Guide 1.183 was used. Calculations performed conclude that expected dose limits following a Fuel handling Accident 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.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment requests: November 7, 2006.

Description of amendment requests: The amendments request to revise Main Steam Safety Valve Requirements and Actions (Technical Specification 3.7.1).

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

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

Response: No.

Based on a detailed plant transient analysis, the Limiting Conditions for Operation (LCOs) and Action statements will continue to restrict operation to within the regions that provide acceptable results. The safety analysis was performed in accordance with the Nuclear Regulatory Commission (NRC) approved San Onofre Units 2 and 3 reload analysis methodology, and considered the concerns identified in NRC Information Notice 94–60.

The increase in Completion Time for Required Action 3.7.1.A.2 from 12 hours to 36 hours is consistent with NUREG–1432 Revision 3, ''Standard Technical Specifications for Combustion Engineering Plants.''

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 add any new equipment, modify any interfaces with any existing equipment, alter the equipment's function, or change the method of operating the equipment. The proposed change does not alter plant conditions in a manner that could affect other plant components. The proposed change does not cause any existing equipment to become an accident initiator.

The increase in Completion Time for Required Action 3.7.1.A.2 from 12 hours to 36 hours is consistent with NUREG–1432 Revision 3, ''Standard Technical Specifications for Combustion Engineering Plants.''

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 Limiting Conditions for Operation (LCOs) and Action statements will continue to restrict operation such that the American Society of Mechanical Engineers (ASME) code requirements continue to be met. The analyses were performed using the NRC approved San Onofre Units 2 and 3 reload analysis methodology. Therefore, the proposed change will have no impact on the margins as defined in the Technical Specification bases.

The increase in Completion Time for Required Action 3.7.1.A.2 from 12 hours to 36 hours is consistent with NUREG–1432 Revision 3, ''Standard Technical Specifications for Combustion Engineering Plants.''

Therefore, this 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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Branch Chief: David Terao.

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

Date of amendment request: November 15, 2006 (TS–459).

Description of amendment request: The proposed amendment requests revision to the Fire Protection License Condition for Units 1, 2, and 3, condition number (13), (14), and (7), respectively, to accommodate operation of Units 1, 2, and 3.

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

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

No. The proposed change revises the license condition to reflect a combined Units 1, 2 and 3 Fire Protection Report. Compliance with the applicable Appendix R requirements is ensured through implementation of the Fire Protection Program and the Appendix R Safe Shutdown Program including Regulatory Issue Summary 2006–10, ''Regulatory Expectations with Appendix R Paragraph III.G.2 Post-Fire Manual Actions.'' The change does not affect any design bases accident or the ability of any safe shutdown equipment to perform its function. Also, although modifications were required to bring BFN in compliance with 10 CFR 50 Appendix R, there are no physical modifications required to implement this license amendment.

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

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

No. The proposed change revises the license condition to reflect a combined Units 1, 2 and 3 Fire Protection Report. Compliance with the applicable Appendix R requirements is ensured through implementation of the Fire Protection Program and Appendix R Safe Shutdown Program including Regulatory Issue Summary 2006–10, ''Regulatory Expectations with Appendix R Paragraph III.G.2 Post-Fire Manual Actions.'' This change does not affect any design basis accident or the ability of any safe shutdown equipment to perform its function. Also, there are no physical modifications required to implement this license amendment. Therefore, this proposed change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

No. The proposed change revises the license condition to reflect a combined Units 1, 2 and 3 Fire Protection Report. Compliance with the applicable Appendix R requirements is ensured through the implementation of the Fire Protection Program and Appendix R Safe Shutdown Program (Units 1, 2, and 3 Fire Protection Report) including Regulatory Issue Summary 2006–10, ''Regulatory Expectations with Appendix R Paragraph III.G.2 Post-Fire Manual Actions.'' The proposed change does not affect any design basis accident and does not reduce or adversely affect the capability to achieve and maintain safe shutdown in the event of a fire. Furthermore, no reductions to the requirements for equipment operability, surveillance requirements or setpoints are being made which could result in reduction in the margin of safety. Therefore, this proposed change will not result in a 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: L. Raghavan.

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.*

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

Date of application for amendments: December 23, 2005, as supplemented by letters dated May 4 and August 3, 2006.

Brief description of amendments: These amendments revise Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' to extend the allowed out of service time for one inoperable emergency diesel generator from 72 hours to 10 days. TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' is revised by the addition of a clarifying note to Condition F of this specification. Additionally, TS 3.4.9, ''Pressurizer,'' is revised to delete the words contained in the limiting condition for operation which require that the two groups of pressurizer heaters are capable of being powered from an emergency power supply.

Date of issuance: December 5, 2006.

Effective date: As of the date of issuance to be implemented within 90 days from the date of issuance.

Amendment Nos.: Unit 1—164, Unit 2—164, Unit 3—164

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating Licenses for all three units.

Date of initial notice in **Federal Register:** January 31, 2006 (71 FR 5080). The May 4 and August 3, 2006, supplemental letters provided additional information that clarified the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 5, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 30, 2004, as supplemented by letters dated March 16, September 29, 2005, and March 21, August 7, August 24, and September 11, 2006.

Brief description of amendment: The license amendment request revised the technical specifications and the final safety analysis report to amend the Columbia Generating Station's licensing and design bases to reflect the application of the alternative source term methodology with an exception. That exception is the Technical Information Document (TID)–14844, ''Calculation of Distance Factors for Power and Test Reactor Sites,'' which will continue to be used as the radiation dose basis for equipment qualification, and radiation zone maps/shielding calculations.

Date of issuance: November 27, 2006.

Effective date: As of its date of issuance and shall be implemented within 120 days from the date of issuance.

Amendment No.: 199.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications and Final Safety Analysis Report.

Date of initial notice in Federal Register: October 26, 2004 (69 FR 62472). The March16, September 29, 2005, and March 21, August 7, August 24, and September 11, 2006, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendment: October 3, 2005.

Brief description of amendment: The amendment revised the Technical Specification (TS) Section 1.1, ''Definitions,'' description of the Pressure and Temperature Limits Report (PTLR), by deleting reference to specifications containing limits in the PTLR; (2) revised administrative controls TS 5.6.6, ''Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),'' by requiring the NRC approval documents to be identified by date and topical reports to be identified by number and title; and (3) added Westinghouse Electric Company, LLC report, WCAP–16143, ''Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,'' to the list of analytical methods provided in TS 5.6.6. The amendment also revises the title of the NRC letter dated August 8, 2001 to clarify the regulation being referenced.

Date of issuance: November 27, 2006.

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

Amendment Nos.: 148, 148, 142, 142.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications and License.

Date of initial notice in Federal Register: March 14, 2006 (71 FR 13175).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 2, 2006, as supplemented by letters dated August 18 and October 5, 2006.

Brief description of amendment: The amendments revised Technical Specification Surveillance Requirement 3.1.7.10, ''Standby Liquid Control System Sodium Pentaborate Isotopic Enrichment'' such that the required enrichment increases from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

Date of issuance: November 16, 2006.

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

Amendment Nos.: 222/214.

Renewed Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specification Surveillance Requirement and Licenses.

Date of initial notice in Federal Register: (71 FR 46931; August 15, 2006).

The August 18 and October 5, 2006, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on August 15, 2006 (71 FR 46931). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 16, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 14, 2005, as supplemented by letter dated September 1, 2006.

Brief description of amendment: The amendment revises Technical Specification (TS) Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' to eliminate the Main Steamline Radiation Monitor trip function.

Date of issuance: November 15, 2006.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 261.

Facility Operating License No. DPR–49: The amendment revises the TSs.

Date of initial notice in Federal Register: (71 FR 43533) August 1, 2006.

The supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination, as published in the **Federal Register** on August 1, 2006 (71 FR 43533).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 22, 2005.

Brief description of amendment: The amendment revises the Duane Arnold Energy Center licensing basis, as described in the Updated Final Safety Analysis Report (UFSAR), to replace the current plant-specific reactor pressure vessel material surveillance program with the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program as the basis for demonstrating compliance with the requirements of Appendix H to Part 50 of Title 10 of the Code of Federal Regulations, ''Reactor Vessel Material Surveillance Program Requirements.''

Date of issuance: November 27, 2006.

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

Amendment No.: 262.

Facility Operating License No. DPR–49: The amendment authorizes changes to the UFSAR.

Date of initial notice in Federal Register: (71 FR 43533) August 1, 2006.

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

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: November 9, 2005, supplemented by letter dated May 15, 2006.

Brief description of amendments: The amendments modify the Technical Specifications (TS) for Prairie Island Nuclear Generating Plant Units 1 and 2, to clarify which TS Surveillance Requirements shall be met for the TS systems which include more components (installed spare components) than are required to satisfy the TS Limiting Conditions for Operation. These amendments revise TS 3.7.8, ''Cooling Water (CL) System,'' TS 3.8.1, ''AC Sources-Operating,'' and TS 3.9.3, ''Nuclear Instrumentation.'' The amendments also make minor corrections for some of these TSs.

Date of issuance: November 14, 2006.

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

Amendment Nos.: 175 and 165.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 2006 (71 FR 7809).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 14, 2006. The supplemental information provided in letter May 15, 2006, did not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the November 9, 2005 submittal.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket No. 50–275, Diablo Canyon Nuclear Power Plant, Unit No. 1, San Luis Obispo County, California

Date of application for amendment: December 16, 2005, as supplemented by a letter dated September 27, 2006.

Brief description of amendment: The amendment revised Technical Specification (TS) 5.6.5, ''Core Operating Limits Report (COLR).'' Specifically, the change added Westinghouse Topical Report WCAP–12945–P–A, Addendum 1–A, Revision 0, ''Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA [Loss-of-Coolant Accident] Evaluation Models,'' dated December 2004, to the list of approved analytical methods in TS 5.6.5.b.

Date of issuance: November 21, 2006.

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

Amendment No.: 191.

Facility Operating License No. DPR–80: The amendment revised the Technical Specifications and Facility Operating License.

Date of initial notice in Federal Register: February 14, 2006 (71 FR 7810).

The September 27, 2006, supplemental letter provided additional information that clarified the application, and did not expand the scope of the application as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 2006.

No significant hazards consideration comments received: No.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of application for amendments: December 6, 2005, as supplemented by letters dated March 14 and November 30, 2006.

Brief description of amendments: The amendment deleted Technical Specification (TS) Limiting Condition for Operation (LCO) 3.3.10, ''Fuel Handling Isolation Signal (FHIS),'' and TS LCO 3.7.14, ''Fuel Handling Building Post-Accident Cleanup Filter System,'' and their associated surveillance requirements. The amendment also deleted the Fuel Handling Building Post-Accident Cleanup Filter Systems from the Ventilation Filter Testing Program in administrative TS 5.5.2.12.

Date of issuance: December 4, 2006.

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

Amendment Nos.: Unit 2—208; Unit 3—200.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Operating Licenses and Technical Specifications.

Date of initial notice in Federal Register: January 3, 2006 (71 FR 155). The March 14 and November 30, 2006, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259 Browns Ferry Nuclear Plant, Unit 1, Limestone County, Alabama

Date of application for amendment: October 12, 2004, as supplemented by letters dated September 7 and November 1, 2006.

Brief description of amendment: To remove License Condition 2.C(4).

Date of issuance: November 28, 2006.

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

Amendment No.: 265

Renewed Facility Operating License Nos. DPR–33: Amendment revised the Renewed Operating License.

Date of initial notice in Federal Register: August 15, 2006 (71 FR 46937). The supplements dated September 7 and November 1, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 28, 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: May 25, 2006, as supplemented by letter dated September 1, 2006.

Brief description of amendments: The requested changes provide a revision to the design and licensing basis for the containment sump debris transport analysis as described in the Sequoyah Nuclear Plant (SQN) Updated Final Safety Analysis Report (UFSAR). The current transport analysis for SQN is a two-dimensional physical transport model, and Tennessee Valley Authority (TVA) is requesting to update the analysis to a three-dimensional computational fluid dynamics transport model. The results of the reanalysis will be used to size the flow area of the advanced design containment sump strainers which will replace the original sump intake structure.

Date of issuance: November 7, 2006.

Effective date: Implementation of the amendment is the incorporation into the next UFSAR update made in accordance with 10 CFR 50.71(e), of the changes to the description of the facility as described in TVA's application dated May 25, 2006, as supplemented by letter dated September 1, 2006, and evaluated in the staff's Safety Evaluation attached to this amendment.

Amendment Nos. 313 and 302.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in Federal Register: June 20, 2006 (71 FR 35460). The supplemental letter dated September 1, 2006, provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 7, 2006.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: September 20, 2006, as supplemented by letter dated November 20, 2006.

Brief description of amendment: The amendment revised (1) the definition of the Pressure and Temperature Limits Report (PTLR) in Technical Specification (TS) 1.1, ''Definitions,'' and (2) TS 5.6.6, ''Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR).''

Date of issuance: December 5, 2006.

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

Amendment No.: 177.

Facility Operating License No. NPF–30: The amendment revised the Operating License and Technical Specifications.

Date of initial notice in Federal Register: October 6, 2006 (71 FR 59136).

The supplemental letter dated November 20, 2006, provided additional clarifying information, 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 published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 5, 2006.

No significant hazards consideration comments received: No.

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.

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

Date of application for amendments: November 9, 2006 (TS–458).

Description of amendments request: The proposed amendment would delete the Technical Specification Surveillance Requirement to verify the position of a low pressure coolant injection crosstie valve.

Date of publication of individual notice in the Federal Register: November 20, 2006 (71 FR 67166).

Expiration date of individual notice: December 20, 2006 (Public comments) and January 19, 2007 (Hearing requests).

Dated at Rockville, Maryland, this 11th day of December 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–21346 Filed 12–18–06; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon written request, copies available from: Securities and Exchange Commission Office of Filings and Information Services, Washington, DC 20549.

Extension: Rule 206(3)–2; SEC File No. 270–216; OMB Control No. 3235–0243.