7778 *Aircraft Accident Report*—Crash During Approach to Landing, Air Tahoma, Inc., Flight 185, Convair 580, N586P, Covington, Kentucky, August 13, 2004.

NEWS MEDIA CONTACT: Ted Lopatkiewicz, Telephone: (202) 314–6100.

Individuals requesting specific accommodations should contact Chris Bisett at (202) 314–6305 by Friday, April 28, 2006.

The public may view the meeting via a live or archived Web cast by accessing a link under ''News & Events'' on the NTSB home page at *http://www.ntsb.gov.*

FOR FURTHER INFORMATION CONTACT: Vicky D'Onofrio, (202) 314–6410.

Dated: April 21, 2006.

Vicky D'Onofrio,

Federal Register Liaison Officer.

[FR Doc. 06–3944 Filed 4–21–06; 1:51 pm]

BILLING CODE 7533–01–M

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meetings

DATE: Weeks of April 24, May 1, 8, 15, 22, 29, 2006.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of April 24, 2006

Monday, April 24, 2006

2 p.m.: Meeting with Federal Energy Regulatory Commission (FERC), FERC Headquarters, 888 First St., NE., Washington, DC 20426, Room 2C (Public Meeting), (Contact: Mike Mayfield, 301–415–3298).

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

Wednesday, April 26, 2006

1 p.m.: Discussion of Management Issues (Closed—Ex. 2).

Thursday, April 27, 2006

1:30 p.m.: Meeting with Department of Energy (DOE) on New Reactor Issues (Public Meeting).

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

Week of May 1, 2006—Tentative

Tuesday, May 2, 2006

9:30 a.m.: Briefing on Status of Emergency Planning Activities—Morning Session (Public Meeting) (Contact: Eric Leeds, 301–415–2334).

1 p.m.: Briefing on Status of Emergency Planning Activities—Afternoon Session (Public Meeting).

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

Wednesday, May 3, 2006

8:55 a.m.: Affirmation Session (Public Meeting) (Tentative). a. ANDREW SIEMASZKO, Docket No. IA–05–021, unpublished Licensing Board Order (Dec. 22, 2005) (Tentative). b. ANDREW SIEMASZKO, Docket No. IA–05–021, unpublished Licensing Board Order (March 2, 2006) (Tentative).

9 a.m.: Briefing on Status of Risk-Informed, Performance-Based Regulation (Public Meeting) (Contact: Eileen McKenna, 301–415–2189).

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

Week of May 8, 2006—Tentative

There are no meetings scheduled for the Week of May 8, 2006.

Week of May 15, 2006—Tentative

Monday, May 15, 2006

1 p.m.: Briefing on Status of Implementation of Energy Policy Act of 2005 (Public Meeting) (Contact: Scott Moore, 301–415–7278).

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

Tuesday, May 16, 2006

9:30 a.m.: Briefing on Results of the Agency Action Review Meeting—Reactors/Materials (Public Meeting) (Contact: Mark Tonacci, 301–415–4045).

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

Week of May 22, 2006—Tentative

Wednesday, May 24, 2006

9:30 a.m.: Discussion of Security Issues (Closed—Ex. 1).

1:30 p.m.: All Employees Meeting (Public Meetings), Marriott Bethesda North Hotel, Salons, D–H, 5701 Marinelli Road, Rockville, MD 20852.

Week of May 29, 2006–Tentative

Wednesday, May 31, 2006

1 p.m.: Discussion of Security Issues (Closed—Ex. 1).

Additional Information

The Briefing on Equal Employment Opportunity (EEO) Programs (Public Meeting) previously scheduled on May 22, 2006, has been postponed and will be rescheduled.

*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.

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

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), 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, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: April 20, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–3945 Filed 4–21–06; 2:01 pm]

BILLING CODE 7590–01–M

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 March 31, 2006 to April 13, 2006. The last biweekly notice was published on April 11, 2006 (71 FR 18371).

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, Rules and Directives 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.*

Dominion Energy Kewaunee, Inc., Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of amendment request: March 17, 2006.

Description of amendment request: The proposed amendment would change the design criteria described in the Kewaunee Power Station (KPS) Updated Safety Analysis Report (USAR). The change would add new design criteria associated with internal flooding to the current licensing basis for KPS.

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

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

Response: No.

The proposed change provides clarification to the existing functional requirements in the USAR by including specific design criteria for analyzing internal flooding in order to verify the capability of an SSC [structure, systems and components] to perform its design function. The proposed change does not affect any of the previously evaluated accidents in the KPS updated safety analysis report (USAR). No SSCs, operating procedures, or administrative controls that have the function of preventing or mitigating any of these accidents are affected.

This proposed change to incorporate design criteria into the USAR provides added administrative assurance that internal flooding will be appropriately addressed, consistent with existing functional requirements, and that safety related SSCs will not be affected by a potential failure of a non-safety related SSC. The change does not affect any accident initiators or the facility accident analysis. Thus, the probability and the consequences of an accident remain unchanged.

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

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

Response: No.

The proposed change to incorporate design criteria consistent with existing functional requirements into the USAR does not change the design function or operation of any safety related SSCs. The proposed change documents design criteria in use and therefore does not involve a physical change to the facility. The change, therefore, does not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

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

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

Response: No.

This proposed change does not affect any margin of safety as established in the Kewaunee USAR because it documents the design criteria presently used and is consistent with the functional requirements in the USAR. This proposed change provides added administrative assurance that safety related SSCs will not be affected by a potential failure of a non-safety related SSC due to a postulated internal flooding event. The proposed change adds criteria for the evaluation of internal flooding events that are more detailed than the existing functional requirements in the USAR. Therefore, the protection and subsequent availability of safety related SSCs is maintained consistent with previously assumed accident mitigation capabilities.

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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Branch Chief: L. Raghavan.

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 amendment request: January 12, 2006.

Description of amendment request: The proposed amendment would correctly modify the wording in Technical Specification Surveillance Requirement (SR) 3.6.6.3 Containment Cooling train cooling water flow rate to accurately reflect the plant configuration. The current SR is to verify flow to each train. The proposed revision to SR 3.6.6.3 would verify flow to each cooler (plant configuration is two coolers per train).

Basis for proposed no significant hazards consideration determination:

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

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

The proposed change will revise Technical Specifications (TS) Surveillance Requirement (SR) 3.6.6.3 containment cooling train cooling water flow rate to accurately reflect the existing plant configuration as described in the Updated Final Safety Analysis Report (UFSAR) Sections 6.2, ''Containment Systems,'' and 9.4, ''Air Conditioning, Heating, Cooling, and Ventilation Systems.'' The revision will specify the appropriate testing requirements for verification that each Containment Cooling System train Essential Service Water (SX) flow rate to each cooling unit is ≥ 2660 gpm [gallons per minute] and will therefore provide assurance that the design flow rate assumed in the safety analyses will be achieved and the Limited Conditions for Operation (LCO) will be met. This change is in the conservative direction, *i.e.*, verification of flow rate to each cooling unit [3] 2660 gpm is more conservative than verification of the same flow rate to each cooling train that consists of two cooling units. The performance of TS surveillance testing is not a precursor to any accident previously evaluated. Thus, the proposed change does not have any effect on the probability of an accident previously evaluated.

The function of the Containment Cooling System in conjunction with the Containment Spray System is to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than design values. There is no change to the design of the Containment Cooling System. Furthermore, the surveillance testing specified in SR 3.6.6.3 will provide assurance that the Containment Cooling System will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not change or 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.

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

Prior to conversion to ITS [Improved Technical Specifications], the SR equivalent to SR 3.6.6.3 required that each system of containment cooling fans be demonstrated OPERABLE by ''verifying an essential service water flow rate of greater than or equal to 2660 gpm to each cooler.'' During the ITS conversion, standard verbiage for SR 3.6.6.3 was adopted; however, the specific plant design of two Reactor Containment Fan Coolers (RCFCs) per Containment Cooling train was inadvertently overlooked.

This proposed amendment would correctly modify the wording in Technical Specifications (TS) Surveillance Requirement (SR) 3.6.6.3 Containment Cooling System to accurately reflect the Braidwood and Byron existing plant design. The revision will provide the appropriate testing requirements for verification that each Containment Cooling System train SX cooling flow rate to each cooling unit is ≥ 2660 gpm. This verification provides assurance that the design flow rate assumed in the safety analyses will be achieved; and, therefore the LCO will be met. The change for verification of SX cooling flow rate from each cooling train to each cooling unit is in the conservative direction and will revise the existing non-conservative TS SR to be consistent with the plant design as described in the UFSAR.

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 requested amendments involve no significant hazards consideration.

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

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: March 23, 2006.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1 (Seabrook) Operating License and Technical Specifications (TSs) to delete the license condition requiring reporting of violations of other requirements (*e.g.*, conditions listed in Section 2.C of the operating license). The change is consistent with the notice published in the **Federal Register** on November 4, 2005, as part of the consolidated line item improvement process.

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

Response: No.

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

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

Response: No.

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

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

Response: No.

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

Based upon the reasoning presented above, 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: M.S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: March 23, 2006.

Description of amendment request: The proposed amendment would revise the Seabrook Station Unit No. 1 (Seabrook) Technical Specifications (TSs) consistent with the NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

Additionally, the proposed amendment would revise Seabrook TS Surveillance Requirement 4.4.6.2.1 to be consistent with NUREG–1431, Revision 3, Improved Standard Technical Specifications Westinghouse Plants.

The NRC staff issued 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 March 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 SGTR [steam generator tube rupture] event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a[n] 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 MSLB [main steamline break], 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[s] to the TS[s] to identify 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 TS[s]. The program, defined by NEI [Nuclear Energy Institute] 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 or 720 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 changes do 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 TS.

Based upon the reasoning presented above, 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: M.S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: Darrell J. Roberts.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: March 7, 2006.

Description of amendment requests: The proposed amendments would modify the Technical Specifications (TS) of the units to change the reactor trip on turbine trip from the P–7 interlock to the P–8 interlock. Specifically, the amendment would effect changes in TS Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' for Function 16, ''Turbine Trip.'' The purpose of the proposed amendment is to decrease potentially unnecessary transients on the reactor and to increase plant availability when the cause of a turbine trip is readily correctable.

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 as follows:

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

Response: No.

The proposed change revises the setpoint at which a reactor trip will occur by changing the interlock at which it is enabled from the P–7 interlock, at approximately 10 percent power, to the P–8 interlock, at less than or equal to 31 percent power. The P–7 and P–8 interlocks are not accident initiators and the change to the reactor trip setpoint does not create any new credible single failure. An analysis has shown that a turbine trip without a reactor trip at 31 percent power or below does not challenge the pressurizer power operated relief valves (PORVs), thereby not adversely affecting the probability of a small[-]break loss[-]of [-]coolant accident due to a stuck open PORV. The consequences of accidents previously evaluated are unaffected by this change because no change to any accident mitigation scenario has resulted and there are no additional challenges to fission product barrier integrity.

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

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

Response: No.

No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed change to the power level at which a reactor trip on turbine trip is enabled does not adversely affect previously identified accident initiators and does not create any new accident initiators. The change does not affect how the associated trip function operates. No new single failures or accident scenarios are created by the proposed change and the proposed change does not result in any event previously deemed incredible being made credible.

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.

No safety analyses [will be] changed or modified as a result of the proposed change in reactor trip setpoint. All margins associated with the current safety analyses acceptance criteria are unaffected. The current safety analyses remain binding. The safety systems credited in the safety analyses will continue to be available to perform their mitigation functions. The proposed change does not affect the availability or operability of safety-related systems and components.

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

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

Attorney for licensee: James M. Petro, Jr., Esquire, One Cook Place, Bridgman, MI 49106.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: January 30, 2006.

Description of amendment request: The proposed change would revise Cooper Nuclear Station (CNS) Technical Specification section 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time extension of no more than 5 years for the Type A, Integrated Leakage Rate Test (ILRT) interval. This revision is a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as defined in Nuclear Energy Institute (NEI) document NEI 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR part 50, appendix J,'' pursuant to 10 CFR 50, appendix J, option B. The requested exception is to allow the ILRT to be performed within 15 years from the last ILRT, last performed on December 7, 1998.

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

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

Response: No.

This license amendment proposes to revise the Technical Specifications to allow for a one-time extension of the ILRT interval from 10 years to 15 years. The containment function is solely to mitigate the consequences of an accident. No design basis accident is initiated by a failure of the containment leakage mitigation function. The extension of the ILRT will not create any adverse interactions with other systems that could result in initiation of a design basis accident. Continued containment integrity is also assured by the established programs for local leakage rate testing and inservice inspections which are unaffected by the proposed change. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from 10 to 15 years. The increase in risk in terms of person-rem per year within 50 miles resulting from accidents was determined to be of a magnitude that NUREG–1493 indicates is imperceptible. NPPD [Nebraska Public Power District] has also analyzed the increase in risk in terms of the frequency of large early releases from accidents. The increase in the large early release frequency resulting from the proposed extension was determined to be within the guidelines published in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NPPD has determined that the increase in conditional containment failure probability from reducing the ILRT frequency from one test in 10 years to one test in 15 years would be insignificant.

Therefore, the probability of occurrence or the consequences of an accident previously analyzed are not significantly increased.

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

Response: No.

The proposed extension of the current interval for the ILRT does not involve any change to the design or operation of any plant structure, system, or component (SSC). The plant will continue to be operated in the same manner. Since no changes to the design or operation of the plant are being made, the proposed one-time extension of the ILRT does not result in a new failure mode for an accident.

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

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

Response: No.

The proposed extension to the ILRT test interval will not result in a change to the design or operation of any plant SSC used to shut down the plant, initiate Emergency Core Cooling Systems, or isolate the primary or secondary containment. Thus, the change will not impact the ability of CNS to mitigate any accident or transient. NUREG–1493, a generic study of the effects of extending containment leakage testing, documented that an extension in the ILRT interval from three per 10 years to one per 20 years resulted in an imperceptible increase in risk to the public. NUREG–1493 generically concluded that the design containment leakage rate contributes about 0.1 percent to the individual risk, and that the decrease in the ILRT frequency would have a minimal effect on this risk since 95% of the potential leakage paths are detected by Type B and Type C testing. A risk assessment using the current CNS Probabilistic Safety Assessment internal events model concluded that the risk associated with this change is very small and not risk significant.

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

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

Attorney for licensee: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

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

Date of amendment request: March 15, 2006.

Description of amendment request: The proposed amendment would revise Cooper Nuclear Station (CNS) Technical Specification 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' by adding two sub-paragraphs to note exemptions from Section III.A and Section III.B of Part 50 of Title 10 of the Code of Federal Regulations, Appendix J, Option B. These two sub-paragraphs allow the leakage contribution from the four main steam line penetrations, referred to as the Main Steam Isolation Valve (MSIV) leakage, to be excluded.

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

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

Response: No.

This proposed change to TS 5.5.12 does not modify existing structures, systems or components (SSC's) of the plant, and it does not introduce new SSC's. It does not change assumptions, methodology or results of previously evaluated accidents in the Updated Safety Analysis Report.

It does not change operating procedures or administrative controls that affect the functions of SSC's. By excluding MSIV leakage from Type A and Type B and C test results, this change will make the CNS Primary Containment Leakage Rate Testing Program more closely aligned with the assumptions used in associated accident consequence analyses. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This proposed change to TS 5.5.12.a does not modify existing SSC's of the plant, and it does not introduce new SSC's. Thus, it does not affect the design function or operation of SSC's involved, and it does not introduce a new accident initiator. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

Since MSIV leakage bypasses the containment and its filtration system (Standby Gas Treatment System) during a Loss-of-Coolant Accident (LOCA), the effects on release to the environment [are] analyzed and specifically accounted for in the CNS dose analysis methodology approved by Amendments 196 and 206. This proposed change to exclude MSIV leakage from Type A and Type B and C test results does not change dose analysis values, and thus, does not affect actual margin in the dose analysis. Therefore, the proposed change does not involve a significant reduction in an actual margin of safety.

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

Attorney for licensee: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

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

Date of amendment request: December 29, 2005.

Description of amendment request: The proposed change would delete Section 2.F of the Nine Mile Point, Unit 2 Facility Operating License (FOL), NPF–69, which requires the licensee report violations of the requirements contained in Section 2.C of this license. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 29, 2005 (70 FR 51098), on possible amendments to delete this reporting requirement, 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 November 4, 2005 (70 FR 67202). The licensee affirmed the applicability of the following NSHC determination in its application dated December 29, 2005.

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

Response: No.

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

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

Response: No.

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

Based on the above, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: March 23, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.4, ''Loss of Power (LOP) Diesel Generator (DG) Start and Load Sequence Instrumentation''. The revision modifies the section title and corrects a nonconservatism in the degraded voltage time delay values in TS Surveillance Requirement (SR) 3.3.4.3.b.

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. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The diesel generators (DGs) provide emergency electrical power to the safeguard

buses in support of equipment required to mitigate the consequences of design basis accidents and anticipated operational occurrences, including an assumed loss of all offsite power. SR 3.3.4.3 verifies that the loss of power (LOP) DG start instrumentation channels respond to measured parameters within the necessary range and accuracy. The proposed amendment revises the section title and corrects nonconservative values in the allowed time delays for the degraded voltage protection function. The revised values are more restrictive than the previously allowed values.

Reducing the time delays for the degraded voltage function as proposed does not significantly increase the probability of a loss of offsite power event. The degraded voltage analysis established both maximum time delay limits for a degraded voltage condition and minimum time delays to prevent premature disconnection from offsite power. The analyzed time delay limits considered prevention of premature disconnection from offsite power such that the probability of an unnecessary loss of offsite power is not significantly increased.

The proposed change does not involve any hardware changes, nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, accident mitigation capabilities, or accident analysis assumptions or inputs.

Therefore, the probability or consequences of any accident previously evaluated will not be significantly increased as a result of the proposed change.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The revised surveillance requirements are more restrictive and will continue to assure equipment reliability such that plant safety is maintained or will be enhanced.

Equipment important to safety will continue to operate as designed. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendment will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The diesel generators (DGs) provide emergency electrical power to the safeguard buses in support of equipment required to mitigate the consequences of design basis accidents and anticipated operational occurrences, including an assumed loss of all offsite power. SR 3.3.4.3 verifies that the loss of power (LOP) DG start instrumentation channels respond to measured parameters within the necessary range and accuracy. The proposed amendment corrects nonconservative values in the allowed time delays for the degraded voltage protection function. The revised values are more restrictive than the previously allowed values. The proposed change to this SR assures that design requirements of the emergency electrical power system continue to be met.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems or components (SSCs) important to safety. Therefore, the requested change will not result in a significant reduction in the margin of safety.

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

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

NRC Branch Chief: L. Raghavan.

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: February 1, 2006.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) requirements for inoperable snubbers by adding Limiting Condition for Operation (LCO) 3.0.8 for SSES 1 and 2. This change is based on the TS Task Force (TSTF) change traveler TSTF–372, Revision 4. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on November 24, 2004, and May 4, 2005.

The Nuclear Regulatory Commission (NRC) staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing license amendment applications in the **Federal Register** on November 24, 2004 (69 FR 68412), and May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the model NSHC determination in its application dated February 1, 2006.

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. 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 delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. Therefore the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. 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). Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, 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 will further minimize possible concerns.

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

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance

of a risk assessment and the management of plant risk. 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 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.

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

Date of amendment request: March 28, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement 3.5.1.4 by changing the method and sample frequency for boron concentration verification for the emergency core cooling system (ECCS) accumulators.

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 ECCS Accumulators are used only to respond to an accident and are not an accident initiator. Therefore, the probability of an accident has not increased.

Boron concentration is controlled in the ECCS Accumulators to prevent either excessive boron concentrations or insufficient boron concentrations. Post-loss-of-coolant accident (LOCA) emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum ECCS Accumulator boron concentration within the upper limit ensures that the ECCS Accumulators do not invalidate these steps. The minimum boron requirements of 2100 (2550 after EPU [extended power uprate]) ppm [parts per million] ppm are based on beginning-of-life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all control element assemblies are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the ECCS Accumulators to prevent a return to criticality during reflood. Level and pressure instrumentation is provided to monitor the availability of the ECCS Accumulators during plant operation.

The Technical Specification Surveillance Requirement (SR 3.5.1.4) verifies that the boron concentration remains within the required range by sampling. Currently, the boron concentration in each ECCS Accumulator is required to be verified by taking a sample of the water in the ECCS Accumulator every 31 days on a staggered test basis. A containment entry is required to take a sample from each of the two ECCS Accumulators. In addition, the makeup water source for the ECCS Accumulators is from the RWST [refueling water storage tank], which is maintained between 2300 ppm and 2600 ppm (2750 and 3050 after EPU) by SR 3.5.4.2, ensuring the ECCS Accumulators are not diluted during makeup/fill evolutions. However, the Reactor Coolant System boron concentration is lower during power operation than the boron concentration in the ECCS Accumulators. Two check valves in series prevent leakage from the Reactor Coolant System into the ECCS Accumulators.

This proposed amendment would require inleakage monitoring to be done every twelve hours in addition to taking samples from each ECCS Accumulator every six months. Samples would continue to be taken to verify the inleakage observations remain conservative.

The engineering analysis and risk insights combine to demonstrate that the method of ECCS Accumulator boron concentration verification can be changed from sampling every 31 days on a staggered test basis to monitoring inleakage every twelve hours and sampling each ECCS Accumulator every six months. The inleakage monitoring is based on a calculational method that has sufficient conservatism to predict the boron concentration of the ECCS Accumulator as shown by sample. Therefore, the ECCS Accumulator would remain capable of responding to an accident as described above and the consequences of an accident previously evaluated are not increased.

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

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 alter the function of any equipment, nor cause it to operate differently than it was designed to operate. All equipment required to mitigate the consequences of an accident would continue to operate as before. The proposed change alters the method of verification of the ECCS Accumulator boron concentration, but not the boron concentration requirements themselves.

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

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

Response: No.

The inleakage monitoring done to verify the concentration of boron in the ECCS Accumulators, is sufficiently conservative to ensure that a decrease in boron concentration would be detected, leading to attempts to increase the boron concentration or a need to sample the affected ECCS Accumulator. Sampling of the ECCS Accumulators every six months will continue to be done to ensure that the inleakage monitoring remains conservative and representative. If the boron concentration is maintained in the ECCS Accumulators, the system operates as assumed in the Updated Final Safety Analysis Report Chapter 15 analyses and the analyses continues to meet the dose consequences acceptance criteria given in the Updated Final Safety Analysis Report.

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

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

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

NRC Branch Chief: Richard J. Laufer.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Houston County, Alabama; Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2, Appling County, Georgia; and Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant (VEGP), Units 1 and 2, Burke County, Georgia

Date of amendment request: February 17, 2006.

Description of amendment request: The proposed amendment would add Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.8 (and renumber existing LCO 3.0.8 to LCO 3.0.9 for VEGP) to allow a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4).

The NRC staff issued a notice of availability of a model safety evaluation and model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on November 24, 2004 (69 FR 68412). The licensee affirmed the applicability of the

model NSHC determination in its application dated February 17, 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change allows a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes 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). Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, 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 will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety. Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a no-significant-hazards consideration.

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

Attorneys for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201; Mr. Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037; Mr. Arthur H. Domby, Troutman Sanders, Nations Bank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Evangelos C. Marinos.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: March 29, 2006.

Description of amendment request: The amendment would revise the Technical Specifications (TS) to 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 proposed amendment includes changes to the TS definition of Leakage; TS 3.4.13, ''Reactor Coolant System, Operational Leakage''; TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program''; and TS 5.6.10, ''Steam Generator Tube Inspection Report''; and adds TS 3.4.17, ''Steam Generator (SG) Tube Integrity.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI (Nuclear Energy Institute) 97–06, ''Steam Generator Program Guidelines.''

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 March 29, 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 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.

Criteria 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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: January 6, 2006 (TS–443).

Description of amendment request: The proposed amendment involves the activation of thermal-hydraulic stability monitoring instrumentation and would allow for the operation of the Oscillating Power Range Monitor (OPRM) module in the ''armed'' mode when the unit returns to power operations. The OPRM module of the Power Range Neutron Monitoring System is designed to provide the licensee's solution regarding reactor stability.

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

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

Response: No

Operating in the region of the power-to-flow map where instabilities can occur may cause a slight, but not significant, increase in the possibility that an instability will occur. This slight increase is acceptable because the OPRM Upscale trip function automatically detects and suppresses design basis thermal-hydraulic power oscillations prior to challenging the fuel MCPR [Minimum Critical Power Ratio] Safety Limit. Thus, the proposed changes do not significantly increase the probability of an accident previously evaluated.

Since the OPRM Upscale trip function precludes challenges to the fuel MCPR Safety Limit, the proposed changes do not involve a significant increase in the 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 changes do not modify the basic functional requirements of the affected equipment nor create any new system failure modes or sequence of events that could lead to an accident. The worst case failure of the affected equipment is failure to perform a mitigation action. Failure of this equipment to perform a mitigating action does not create the possibility of a new or different kind of accident.

No new external threats or release pathways are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed changes do not revise any safety margin requirements. The OPRM Upscale trip function is designed to meet all requirements of General Design Criteria (GDC) 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic power oscillations prior to challenging the fuel MCPR Safety Limit. Thus, the new equipment improves the ability of the equipment to automatically enforce compliance with margins of safety.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Michael L. Marshall, Jr.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: February 24, 2006 (TS–06–02).

Description of amendment request: The proposed amendment would revise the Updated Final Safety Analysis Report (UFSAR) Section 15.5 dose analysis inputs and results for the steam generator tube rupture (SGTR) accident. This analysis is being revised for both the current steam generators and the revised primary and secondary side

mass releases associated with the new replacement steam generators, which are scheduled to be installed during the Unit 1, Cycle 7 Refueling Outage in the Fall 2006. The analysis for the current steam generators was revised as a result of an error identified in the computer model used to calculate the dose consequences to the Main Control Room subsequent to an accident.

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 postulated SGTR analysis was revised to determine the control room operator and offsite dose due to correction of computer model input errors and for primary and secondary side mass releases associated with the replacement steam generators. The COROD and Control Room Emergency Ventilating System (CREVS) computer model input errors are software issues which affect analysis results but do not affect operation of plant systems. Consequently, correction of these errors does not have an affect on the probability of occurrence of an accident. The change in the primary and secondary side mass releases associated with the replacement steam generators results in changes to the input to the current SGTR accident analysis. The revised analysis results in an increase the calculated Main Control Room (MCR) SGTR doses. However, the changes in primary and secondary side mass releases and associated release time sequence does not increase the probability of an accident previously evaluated.

The COROD and CREVS computer model input errors and revised primary and secondary side mass releases associated with the replacement steam generators will result in an increase in the calculated MCR pre-accident iodine spike thyroid dose; however the resulting calculated MCR dose does not exceed 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, ''Control Room,'' dose limits as specified in NUREG–0800, ''Standard Review Plan.'' Other offsite and MCR doses (gamma, beta, and thyroid) associated with the SGTR accident for the current steam generators and the replacement steam generators either remain the same, decrease slightly or increase slightly. These changes are within the ten percent allowable increase criteria of NEI [Nuclear Energy Institute] 96–07, Revision 1. These doses remain within a small fraction of the 10 CFR 100, ''Reactor Site Criteria,'' and 10 CFR 50 Appendix A, GDC 19 as specified in NUREG–0800. Consequently, the changes do not involve a significant increase in the consequences of an accident previously evaluated.

Based on the above, 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 COROD and CREVS computer model input errors are software issues which affect analysis results but do not result in new accident initiators since operation of plant systems and equipment are not affected. Thus, these input changes do not create the possibility of new or different kind of accident from those previously evaluated. The change in the primary and secondary side mass releases associated with the replacement steam generators result in changes to the input to the current SGTR accident analysis. The revised analysis results in an increase in the calculated MCR doses. However, the changes in primary and secondary side mass releases and associated release time sequence do not create the possibility of a new or different kind of accident than previously evaluated.

Based on the above, the changes will not initiate an accident nor create any new failure mechanisms. The changes do not result in any event previously deemed incredible being made credible. In addition; the changes will not result in any increase in the challenges to safety systems. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed changes to the affected UFSAR tables revise the calculation input for offsite and MCR dose values for the SGTR accident. The MCR thyroid dose (21 μCi/gm case) for the current steam generators and the revised mass releases associated with the replacement steam generators exceeds the ten percent allowable increase criteria of NEI 96–07, Revision 1. Offsite doses for the current steam generators remain the same and then decrease slightly for the replacement steam generators. The MCR gamma and beta doses (21 μCi/gm case) increase slightly for the current steam generators and then decrease slightly for the replacement steam generators. The MCR gamma, beta and thyroid doses (0.265 μCi/gm case) increase slightly for the current steam generators and then decrease slightly for the revised mass releases associated with the replacement steam generators.

The above changes in SGTR accident doses are acceptable since the MCR doses do not exceed the requirements in 10 CFR 50, Appendix A, GDC 19 and the whole body and thyroid doses at the exclusion area and the lower population zone outer boundaries remain the same or decrease relative to the UFSAR values. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902

NRC Branch Chief: Michael L. Marshall, Jr.

Notice of Issuance of Amendments To Facility Operating Licenses

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

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

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

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

located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendments: August 11, 2005, as supplemented by letters dated October 11, November 16, and December 12, 2005, and February 7, 2006.

Brief Description of amendments: The amendments revise Technical Specification (TS) Surveillance Requirement 3.6.1.3.9 with respect to the allowed leakage rate through each Main Steam Isolation Valve.

Date of issuance: March 2, 2006.

Effective date: March 2, 2006.

Amendment Nos.: 239 and 267.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the TS.

Date of initial notice in ***Federal Register:*** September 13, 2005 (70 FR 54087). The letters dated October 11, November 16, and December 12, 2005, and February 7, 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 March 2, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 6, 2005, as supplemented by letters dated August 8, and December 9, 2005.

Brief description of amendment: This amendment revises Technical Specification (TS) 6.8.4.k, ''Containment Leakage Rate Testing Program'' and TS Surveillance Requirement 4.6.1.6.1, ''Containment Vessel Surfaces.'' Specifically, the amendment allows a one-time extension of Appendix J to Part 50 of Title 10 of the Code of Federal Regulation, Type A, Containment Integrated Leak Rate Test interval from once in 10 years to once in 15 years.

Date of issuance: March 30, 2006.

Effective date: March 30, 2006.

Amendment No.: 122.

Facility Operating License No. NPF–63: Amendment revises the TS.

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59084). The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 30, 2006.

No significant hazards consideration comments received: No.

Dairyland Power Cooperative, Docket No. 50–409, La Crosse Boiling Water Reactor, Genoa, Wisconsin

Date of amendment request: December 13, 2005.

Brief description of amendment: The amendment revises Technical Specifications to allow waste processing components or fixtures to be handled over the Fuel Element Storage Well (FESW), limiting the weight of such items to 50 tons (the weight of the heavy load drop found acceptable in the cask drop analyses performed for the La Crosse Boiling Water Reactor FESW).

Date of issuance: April 3, 2006.

Effective date: April 3, 2006.

Amendment No.: 70.

Possession Only License No. DPR–45: The amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 14, 2006 (71 FR 7804).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation Report, dated April 3, 2006.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: May 24, 2005.

Brief description of amendment: The amendment revised the applicability requirements of Technical Specification 3.7.A.5.a. and 3.7.A.i. related to primary containment oxygen concentration and drywell-to-suppression chamber differential pressure limits.

Date of issuance: April 10, 2006.

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

Amendment No.: 218.

Facility Operating License No. DPR–35: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 30, 2005 (70 FR 51380).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 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; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: June 15, 2005, as supplemented by letters dated January 26, January 31, February 22, March 3, and March 23, 2006.

Brief description of amendments: The amendment allows a transition to Westinghouse SVEA–96 Optima2 fuel at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) beginning with the QCNPS, Unit 2 refueling outage in March 2006. Specifically, the amendment revised Technical Specifications (TSs) Section 3.1.4, ''Control Rod Scram Times,'' TS Section 4.2.1, ''Fuel Assemblies,'' and TS Section 5.6.5, ''Core Operating limits Report (COLR),'' to support this transition. Additionally, a new surveillance requirement was added to verify sodium pentaborate enrichment. The core reload analyses using the new Westinghouse analytical methods for the affected units may result in the need for additional TS changes to support the transition to Westinghouse SVEA–96 Optima2 fuel, such as a change to the safety limit minimum critical power ratio.

Date of issuance: April 4, 2006.

Effective date: As of the date of issuance and shall be implemented prior to unit startup with a reactor core containing Westinghouse SVEA–96 Optima2 fuel.

Amendment Nos.: 220/211, 231/227.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30. The amendments revised the Technical Specifications and Surveillance Requirements.

Date of initial notice in ***Federal Register:*** July 19, 2005 (70 FR 41445).

The January 26, January 31, February 22, March 3, and March 23, 2006, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 2006. No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–265, Quad Cities Nuclear Power Station, Unit 2, Rock Island County, Illinois

Date of application for amendments: December 15, 2005, as supplemented by letters dated February 13 and March 3, 2006.

Brief description of amendments: The amendment revised the safety limit minimum critical power ratio values in Technical Specification (TS) Section 2.1.1, ''Reactor Core SLs.'' Specifically, the change required that for Quad Cities, Unit 2, the minimum critical power ratio (MCPR) for Global Nuclear Fuel fuel shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation. Additionally, the change required that the MCPR for Westinghouse fuel shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.13 for single loop operation.

Date of issuance: March 31, 2006.

Effective date: As of the date of issuance and shall be implemented prior to unit startup with a reactor core containing Westinghouse Optima2 fuel.

Amendment No.: 226.

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

Date of initial notice in ***Federal Register:*** January 17, 2006 (71 FR 2591).

The February 13, 2006, and March 3, 2006, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 2006.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: November 8, 2004, as supplemented March 31, 2005, and February 13, 2006.

Brief description of amendment: The amendment revises Technical Specification (TS) Section 4.4.5.4 to modify the definitions of steam generator tube ''Plugging Limit'' and ''Tube Inspection.'' The purpose of these modifications is to define the depth of the required tube inspections and to clarify the plugging criteria within the tubesheet region. The amendment also modifies TS Section 4.4.5.5, ''Reports,'' to require a Special Report of indications found in the tubesheet region following each inspection.

Date of Issuance: April 11, 2006.

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

Amendment No.: 143.

Renewed Facility Operating License No. NPF–16: Amendment revised the TS.

Date of initial notice in ***Federal Register:*** November 24, 2004 (69 FR 68404).

The March 31, 2005, and February 13, 2006, Supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 13, 2005, as supplemented by letter dated September 29, 2005.

Brief description of amendment: The amendment incorporated several Technical Specification Task Force (TSTF) changes to the licensee's Technical Specifications (TSs). The specific TSTF changes that were incorporated are:

1. TSTF–222–A, Revision 1, ''Control Rod Scram Time Testing''—This change modifies TS Section 3.1.4, ''Control Rod Scram Times,'' to clarify that control rod scram time testing is required only for core cells in which work on the control rod or drive has been performed or fuel has been moved or replaced.

2. TSTF–275–A, Revision 0, ''Clarify Requirement for EDG [emergency diesel generator] start signal on RPV [reactor pressure vessel] Level—Low, Low, Low during RPV cavity flood-up''—This change modifies the TS Section 3.3.5.1, ''ECCS [emergency core cooling system] Instrumentation,'' to clarify that the ECCS initiation instrumentation, identified as being required in modes 4 and 5, is required to be operable only when the associated ECCS subsystems are required to be operable as defined in limiting condition of operation (LCO) 3.5.2, ''ECCS—Shutdown.''

3. TSTF–300–A, Revision 0, ''Eliminate DG [diesel generator] LOCA [loss-of-coolant accident]-Start SRs [surveillance requirements] while in S/D [shutdown] when no ECCS is Required''—This change modifies the TS Section 3.8.2, ''AC [alternating current] Sources—Shutdown,'' to add an additional note to the surveillance that verifies automatic start of the emergency diesel generators and automatic load shedding from the emergency buses, is considered to be met without the ECCS initiation signals operable when ECCS initiation signals are not required to be operable per Table 3.3.5.1–1, ECCS Instrumentation.

4. TSTF–225, Revision 2, ''Fuel movement with inoperable refueling equipment interlocks''—This change modifies TS Section 3.9.1, ''Refueling Equipment Interlocks,'' to add required actions to allow insertion of a control rod withdrawal block and verification that all control rods are fully inserted as alternate actions to suspending in-vessel fuel movement in the event that one or more required refueling equipment interlocks are inoperable.

Date of issuance: March 30, 2006.

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

Amendment No.: 218.

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

Date of initial notice in ***Federal Register:*** June 7, 2005 (70 FR 33216).

The supplement dated September 29, 2005, 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 March 30, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 8, 2005, as supplemented by letters dated March 17 and 27, 2006.

Brief description of amendment: The amendment adds limits and controls for the spent fuel cask loading and unloading operations in the spent fuel pool (SFP). The change modifies the technical specifications (TSs) by adding a new Limiting Condition for Operation (LCO) 2.8.3(6) that establishes (1) A boron concentration requirement during cask loading operations in the SFP, and (2) a spent fuel burnup-initial enrichment limit in the spent fuel cask to ensure subcritical conditions are maintained during spent fuel cask loading operations in the SFP. In addition, the change modifies TS Tables 3–4 and 3–5, and adds a new subsection 4.3.1.3 in Design Features 4.3.1 to describe the spent fuel cask design

features. In addition, editorial changes were made mostly to make the TSs consistent with the proposed changes and to conform pagination.

Date of issuance: April 10, 2006.

Effective date: The license amendment is effective as of its date of issuance.

Amendment No.: 239.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 20, 2005 (70 FR 75494).

The March 17 and 27, 2006, supplemental letters provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a safety evaluation dated April 10, 2006.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of application for amendments: October 5, 2005.

Brief description of amendments: The amendments change the SSES 1 and 2 Technical Specifications (TSs) 3.4.10, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits,'' by removing the valid P/T curve limit date and replacing it with the effective full-power years (EFPY) of radiation exposure on each of the P/T limit curves for SSES 1 and 2. The new P/T limit will be 35.7 EFPY for SSES 1 and 30.2 EFPY for SSES 2.

Date of issuance: March 30, 2006.

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

Amendment Nos.: 232 and 209.

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

Date of initial notice in ***Federal Register:*** January 17, 2006 (71 FR 2595).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 30, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 5, 2005, as supplemented on March 31, 2006.

Brief description of amendments: These amendments revise the Technical Specifications by eliminating the requirements to submit monthly operating reports and occupational radiation exposure reports.

Date of issuance: April 6, 2006.

Effective date: April 6, 2006.

Amendment Nos.: 233 and 210.

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

Date of initial notice in ***Federal Register:*** January 3, 2006 (71 FR 153).

The supplement dated March 31, 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.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 6, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 5, 2005, as supplemented on March 31, 2006.

Brief description of amendments: These amendments revise the Technical Specifications by eliminating the requirements associated with hydrogen recombiners, and hydrogen and oxygen monitors.

Date of issuance: April 6, 2006.

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

Amendment Nos.: 234 and 211.

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

Date of initial notice in ***Federal Register:*** January 3, 2006 (71 FR 152).

The supplement dated March 31, 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.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 6, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 11, 2005.

Brief description of amendment: The amendment revises certain 18-month Technical Specification (TS) surveillance requirements to eliminate the condition that testing be conducted during shutdown conditions.

Date of issuance: April 4, 2006.

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

Amendment No.: 165.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** January 17, 2006 (71 FR 2593).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 11, 2005.

Brief description of amendment: The amendment removes the Technical Specification (TS) 3.1.5 requirement for the standby liquid control (SLC) system to be operable in Operational Condition 5 (refueling) with any control rod withdrawn. Corresponding changes are also made to the SLC initiation sections of TS Tables 3.3.2–1 and 4.3.2–1.

Date of issuance: April 7, 2006.

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

Amendment No.: 166.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** January 31, 2006 (71 FR 5083).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 11, 2005.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) to relocate the component identification of the

overcurrent protective devices from TS 3/4.8.4.1 and TS 3/4.8.4.5 to the Updated Final Safety Analysis Report.

Date of issuance: April 10, 2006.

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

Amendment No.: 167.

Facility Operating License No. NPF–57: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** March 6, 2006 (71 FR 11233).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 2006.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: September 27, 2005.

Brief Description of amendments: The amendments revise the Technical Specifications to eliminate the power range neutron high-flux negative rate reactor trip function.

Date of issuance: February 27, 2006.

Effective date: As of the date of issuance and shall be implemented prior to startup following refueling outage 21 for Unit 1 and prior to startup following refueling outage 18 for Unit 2.

Amendment Nos.: 171 and 164.

Renewed Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 8, 2005 (70 FR 67750).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: August 30, 2005.

Brief description of amendments: The amendments revise Technical Specifications to reflect incorporation of the Westinghouse Electric Company Best Estimate Analyzer for Core Operations—Nuclear power distribution monitoring as described in Topical Report WCAP–124–P–A, ''BEACON—Core Monitoring and Operations Support System.''

Date of issuance: March 31, 2006.

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

Amendment Nos.: Unit 1–175; Unit 2–163.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications and Surveillance Requirements.

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59088).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 30, 2005.

Brief description of amendments: The amendments revise Technical Specifications to reflect incorporation of the Westinghouse Electric Company Best Estimate Analyzer for Core Operations—Nuclear power distribution monitoring as described in Topical Report WCAP–124–P–A, ''BEACON—Core Monitoring and Operations Support System.''

Date of issuance: March 31, 2006.

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

Amendment Nos.: Unit 1–175; Unit 2–163.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications and Surveillance Requirements.

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59088).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 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: September 1, 2005, as supplemented by letters dated March 16 and 30, 2006.

Brief description of amendments: The amendments temporarily revise the reactor protection system turbine trip allowable value for low trip system pressure from greater than or equal to 43 pounds per square inch gauge (psig) to 39.5 psig for Operating Cycle 15.

The amendments revise Technical Specification 2.2.1, Functional Unit 17.A allowable value in Table 2.2–1 ''Reactor Trip System Instrumentation Setpoints.''

Date of issuance: April 6, 2006.

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

Amendment Nos. 307 and 296.

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

Date of initial notice in ***Federal Register:*** October 25, 2005 (70 FR 61662). The supplemental letters provided clarifying information that 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 April 6, 2006.

No significant hazards consideration comments received: No.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: April 13, 2004, as supplemented by letters dated March 18 and August 31, 2005, and January 6, 2006.

Description of amendment: The amendments revise the Technical Specification (TS) 3.3.2, ''Engineered Safety Features Actuation System Instrumentation, '' Function 7.b, ''Refueling Water Storage Tank Level—Low Low'' trip setpoint, and revise the frequency of calibration of the level transmitters from every 9 months to every 18 months.

Date of issuance: March 30, 2006.

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

Amendment Nos.: 125 and 125.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications and Surveillance Requirements.

Date of initial notice in ***Federal Register:*** May 11, 2004 (69 FR 26193). The March 18 and August 31, 2005, and January 6, 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 proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 30, 2006.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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

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

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, 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, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert

opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1] Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

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

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

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

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for 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 or 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).

AmerGen Energy Company, Docket No. 50–289, Three Mile Island, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: April 6, 2006.

Description of amendment request: The amendment revised Technical Specification (TS) 3.7.2.c, ''Unit Electric Power System,'' to increase the TS allowed outage time with one inoperable emergency diesel generator EDG–Y–1A from 7 days to 10 days, on a one-time basis.

Date of issuance: April 8, 2006.

Effective date: As of the date of issuance and is applicable until the emergency diesel generator EG–Y–1A is returned to operable status or until April 12, 2006, at 21:00 hours, whichever occurs first.

Amendment No.: 258.

Facility Operating License No. DPR–50: The amendment revised the TSs.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, State consultation, and final NSHC determination are contained in a safety evaluation dated April 8, 2006.

Attorney for licensee: Assistant General Counsel, AmerGen Energy Company, LLC 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Darrell J. Roberts.

Arizona Public Service Company, et al., Docket No. STN 50–528, Palo Verde Nuclear Generating Station, Unit No. 1, Maricopa County, Arizona

Date of application for amendment: March 31, 2006, as supplemented by letters dated March 31 and April 4, 2006.

Brief description of amendment: The amendment to the Updated Final Safety Analysis Report allows the use of an operator action as a compensatory measure to prevent exceeding the Train A shutdown cooling (SDC) system design basis vibration limit if a Loop 2 reactor coolant pump (RCP) should trip or have a sheared shaft during four-RCP operation. This compensatory measure would only be used during a one-time 12-hour period for root cause data collection in Mode 3. After the root cause data collection is completed, a modification will be implemented to reduce the SDC system vibration.

Date of issuance: April 6, 2006.

Effective date: April 6, 2006, and shall be implemented within 5 days of the date of issuance.

Amendment No.: Unit 1–159.

Facility Operating License No. NPF–41: The amendment revises the Updated Final Safety Analysis Report as set forth in the application for amendment by licensee letter dated March 31, 2006, as supplemented.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. A public notice was published in the April 3 and 4, 2006, editions of the Arizona Republic. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC

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

determination are contained in a safety evaluation dated April 6, 2006. The March 31 and April 4, 2006, supplemental letters provided additional clarifying information, 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.

Attorney for licensee: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Branch Chief: David Terao.

Florida Power and Light, et al., Docket No. 50–389, St. Lucie Nuclear Plant, Unit 2, St. Lucie County, Florida

Date of amendment request: February 21, 2006.

Description of amendment request: The amendment revises the Technical Specifications (TSs) for the Containment Ventilation System to allow additional corrective actions for inoperable containment purge supply and exhaust valves. These corrective actions are consistent with the Standard TSs for Combustion Engineering plants.

Date of issuance: March 17, 2006.

Effective date: March 17, 2006.

Amendment No.: 142.

Facility Operating License No. NPF–16: Amendment revises the TSs.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. 71 FR 10566 dated March 1, 2006. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by May 1, 2006, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated March 17, 2006.

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

NRC Branch Chief: Michael L. Marshall, Jr.

Southern Nuclear Operating Company, Inc., Docket No. 50–425, Vogtle Electric Generating Plant, Unit 2, Burke County, Georgia

Date of amendment request: March 29, 2006.

Description of amendment request: The amendment revised TS 3.7.6, ''Condensate Storage Tank (CST),'' to require two CSTs to be OPERABLE and to increase the combined safety-related minimum volume. The amendment also revised Surveillance Requirement 3.7.6 to reflect the additional limit for CST volume. This amendment is needed to resume power operation at the Vogtle Electric Generating Plant, Unit 2.

Date of issuance: March 31, 2006.

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

Amendment No.: 120.

Facility Operating License No. NPF–81: Amendment revises the technical specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, State consultation, and final NSHC determination are contained in a safety evaluation dated March 31, 2006.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Evangelos C. Marinos.

Dated at Rockville, Maryland, this 17th day of April 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. 06–3901 Filed 4–24–06; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE

Board of Governors; Sunshine Act Meeting

TIMES AND DATES: 10:30 a.m., Tuesday, May 2, 2006; 8:30 a.m. and 10 a.m., Wednesday, May 3, 2006.

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

STATUS: May 2, 10:30 a.m. (Closed); May 3, 8:30 a.m. (Open); May 3, 10 a.m. (Closed).

MATTERS TO BE CONSIDERED:

Tuesday, May 2, at 10:30 a.m. (Closed)

1. Strategic Planning.
2. Financial Update.
3. Rate Case Planning.
4. Labor Negotiations Planning.
5. Personnel Matters and Compensation Issues.

Wednesday, May 3, at 8:30 a.m. (Open)

1. Minutes of the Previous Meetings, February 7–8; and March 22–23, 2006.
2. Remarks of the Postmaster General and CEO Jack Potter.
3. Committee Reports and Committee Charters.
4. Capital Investments.
   a. Automated Flat Sorting Machine 100—Auto Induction Phase 2.
   b. Additional Delivery Barcode Sorter Equipment.
   c. Oklahoma City, Oklahoma, Regional Distribution Center.
5. Quarterly Report on Service Performance.
6. Quarterly Report on Financial Performance.
7. 2006 Privacy Trust Study of the U.S. Government.
8. Tentative Agenda for the June 6–7, 2006 meeting in Indianapolis, Indiana.

Wednesday, May 3 at 10 a.m. (Closed)—(If Needed)

1. Continuation of Tuesday's closed session agenda.

FOR FURTHER INFORMATION CONTACT: Wendy A. Hocking, Secretary of the Board, U.S. Postal Service, 475 L'Enfant Plaza, SW., Washington, DC 20260–1000. Telephone (202) 268–4800.

Wendy A. Hocking,

Secretary.

[FR Doc. 06–3950 Filed 4–21–06; 3:32 pm]

BILLING CODE 7710–12–M

RAILROAD RETIREMENT BOARD

Actuarial Advisory Committee With Respect to the Railroad Retirement Account; Notice of Public Meeting

Notice is hereby given in accordance with Public Law 92–463 that the Actuarial Advisory Committee will hold a meeting on May 24, 2006, at 10 a.m. at the office of the Chief Actuary of the U.S. Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois, on the conduct of the 23rd Actuarial Valuation of the Railroad Retirement System. The agenda for this meeting will include a discussion of the results and presentation of the 23rd Actuarial Valuation. The text and tables which constitute the Valuation will have prepared in draft form for review by the Committee. It is expected that this will be the last meeting of the Committee before publication of the Valuation.

The meeting will be open to the public. Persons wishing to submit written statements or make oral