describes the function of the Board. Notice of the meeting is required under the Sunshine in Government Act.

TIME AND DATE: Monday, June 5, 2006 from 10:30 a.m. to 12 p.m.

AGENDA: Committee Meetings of the Eighth National Museum and Library Service Board Meeting:

10:30 a.m.–12 p.m. Joint Meeting of the Committees on Partnerships & Government Affairs and the Committee on Policy & Planning.
(Open to the Public)
I. Staff Reports.
II. Other Business.
2 p.m.–3:30 p.m. Jury Meeting to consider the National Awards for Museum Services.
(Closed to the Public)
4 p.m.–5:30 p.m. Jury Meeting to consider the National Awards for Library Services.
(Closed to the Public)

PLACE: The meetings will be held at the Institute of Museum and Library Services, 1800 M Street, NW., 9th Floor, Washington, DC 20036. Telephone: (202) 653–4676.

TIME AND DATE: Tuesday, June 6, 2006, from 9 a.m. to 1 p.m.

AGENDA: Eighth National Museum and Library Services Board Meeting: (Open to the Public)

I. Welcome.
II. Approval of Minutes.
III. Program Reports.
IV. Committee Reports.
V. Board Program: Big Read Initiative.
VI. Other Business.
VII. Adjournment.

PLACE: The meeting will be held at the Institute of Museum and Library Services, 1800 M Street, NW., 9th Floor, Washington, DC 20036. Telephone: (202) 653–4676.

STATUS: Parts of this meeting will be closed to the public as identified in the meeting agenda and **SUPPLEMENTARY INFORMATION.** The rest of the meeting will be open to the public.

FOR FURTHER INFORMATION CONTACT: Elizabeth Lyons, Special Assistant to the Director, Institute of Museum and Library Services, 1800 M Street, NW., 9th Floor, Washington, DC 20036. Telephone: (202) 653–4676.

SUPPLEMENTARY INFORMATION: The National Museum and Library Services Board is established under the Museum and Library Services Act, 20 U.S.C. 9101 *et seq.* The Board advises the Director of the Institute on general policies with respect to the duties, powers, and authorities related to Museum and Library Services.

The Jury Meetings to Consideration the National Awards for Museum and Library Services, on Monday, June 5, 2006, will be closed pursuant to subsections (c)(4) and (c)(9) of section 552b of Title 5, United States Code because the Board will consider information that may disclose: Trade secrets and commercial or financial information obtained from a person and privileged or confidential; and information the premature disclosure of which would be likely to significantly frustrate implementation of a proposed agency action. The meetings from 10:30 a.m. until 12 p.m. on Monday, June 5, 2006 and the meeting from 9 a.m. to 1 p.m. on Tuesday, June 6, 2006, are open to the public. If you need special accommodations due to a disability, please contact: Institute of Museum and Library Services, 1100 Pennsylvania Avenue, NW., Washington, DC 20506. Telephone: (202) 653–4676; TDD (202) 653–4699 at least seven (7) days prior to the meeting date.

Dated: May 17, 2006.

Kate Fernstrom,

Chief of Staff.

[FR Doc. 06–4804 Filed 5–19–06; 10:22 am]

BILLING CODE 7036–01–M

NATIONAL TRANSPORTATION SAFETY BOARD

Notice of Sunshine Act Meeting

TIME AND DATE: 9:30 a.m., Wednesday, May 31, 2006.

PLACE: NTSB Conference Center, 429 L'Enfant Plaza, SW., Washington, DC 20594.

STATUS: This one item is open to the public.

MATTER TO BE CONSIDERED: 7794, *Highway Accident Brief*—Passenger Vehicle Collison with a Fallen Overhead Girder Eastbound on Interstate 70 at the Colorado State Route 470 Overpass, Golden, Colorado, May 15, 2004.

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

Individuals requesting specific accommodations should contact Chris Bisett at (202) 314–6305 by Friday, May 26, 2006.

The public may view the meeting via a live or archived Webcast 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: May 19, 2006.

Vicky D'Onofrio,

Federal Register Liaison Officer.

[FR Doc. 06–4836 Filed 5–19–06; 2:41 pm]

BILLING CODE 7533–01–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 April 28, 2006 to May 11, 2006. The last biweekly notice was published on May 9, 2006 (71 FR 26995).

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: April 6, 2006.

Description of amendment request: The proposed amendment would allow the use of a different methodology for determining the design requirements necessary for protecting safety-related equipment from damage by tornado generated missiles.

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 probability of occurrence of an accident previously evaluated is not significantly increased by the proposed change to permit probabilistic evaluation of missiles generated by natural phenomena. The actual frequency of tornado occurrence at Kewaunee is unaffected by the proposed change in assessment methodology. Furthermore, the projected frequency of tornado occurrence, as specified in the USAR [Updated Safety Analysis Report], is not significantly affected by this change. The value for the probability of tornado occurrence in the updated study is in general agreement with the original value in the USAR (i.e. 3.97E–4 vs. 4.86E–4). Similarly, the probability of a tornado-generated missile is not significantly affected by this change.

Likewise, the consequences of an accident previously evaluated are not significantly increased by the proposed change. The actual probability of a tornado missile onsite remains unchanged. The actual probability of a tornado missile strike remains unchanged. For the limited number of components affected by this proposed change (i.e. exhaust ducts and fuel vent), the missile strike probability is approximately 5.75 x 10^{-7} per year, which is significantly lower than the SRP [Standard Review Plan] acceptance criteria of 1 x 10^{-6} per year. Therefore, the proposed change is not considered to constitute a significant increase in the consequences of an accident due to the low probability of occurrence.

In addition, use of a probabilistic versus a deterministic methodology to assess missile hazard acceptability has no impact on accident initiation or consequence. 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 changes involve use of an evaluation methodology to determine protection requirements for two specific support components for safety-related equipment, which may be adversely affected by missiles during a tornado. A tornado at Kewaunee is considered in the USAR as a separate event and not occurring coincident with any of the design basis accidents in the USAR. As such, no new or different kind of accident is created by the proposed change to permit probabilistic evaluation of missiles generated by natural phenomena.

This change involves recognition of the acceptability of performing tornado missile strike probability calculations in accordance with established regulatory guidance in lieu of using deterministic methodology alone. Therefore, the change would not create the possibility of, or be the initiator for, any new or different kind of accident. The acceptance criterion of the SRP guidance establishes a threshold for tornado missile damage to system components that is consistent with this conclusion.

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.

The request does not involve a significant reduction in a margin of safety. The existing design basis for Kewaunee, with respect to a tornado affecting safety related equipment, is to provide positive missile barriers for all safety-related systems and components. The proposed change recognizes that for probability of occurrences below the SRP established acceptance limit, the extremely low probability associated with an "important" system or component being struck by a tornado missile does not represent a significant reduction in the margin of safety provided by use of the deterministic methodology. The change from "protecting all safety-related systems and components" to "an extremely low probability of occurrence of tornado generated missile strikes on portions of important systems and components" is not considered to constitute a significant reduction in the margin of safety.

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.

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

Date of amendment request: March 28, 2006.

Description of amendment request: The proposed amendment would eliminate redundant surveillance requirements [SRs] pertaining to post-maintenance/post-modification testing. The associated TS bases will be updated to address the proposed changes.

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 changes do not modify any plant equipment and do not impact any failure modes that could lead to an accident. Testing in accordance with the requirements of SR 4.0.1 will continue to provide the necessary assurance that the associated systems will function consistent with the assumptions used in the accident analyses. On this basis, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not involve any physical changes to systems, structures, or components, or involve a change to the method of plant operation. The requirement to perform post maintenance/post modification testing will continue to be implemented consistent with SR 4.0.1, through existing plant programs and procedures. As such, the proposed amendment does not introduce any new failure modes, accident initiators or malfunctions that would cause a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The TS changes do not involve a significant reduction in a margin of safety because the requirements described in SR 4.0.1, as implemented through existing plant programs and procedures, will continue to ensure that post maintenance/post modification testing will be performed when necessary. The proposed change does not affect any of the assumptions used in the accident analyses, nor does it affect operability requirements for equipment important to plant safety. Therefore, the margin of safety is not impacted by the proposed amendment.

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: April 17, 2006.

Description of amendment request: The proposed amendment would change the method for calculating fuel pool decay heat load from the original licensing basis methodology of ORIGEN and the Auxiliary Systems Branch Technical Position (ASBTP) 9–2, ''Residual Decay Heat Energy for Light Water Reactors for Long-Term Cooling,'' to ORIGEN–ARP.

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 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 adoption of ORIGEN–ARP does not affect the probability or consequences of an accident previously evaluated. The calculation of the fuel pool decay heat load is used to evaluate and demonstrate the ability of the fuel pool cooling system to maintain the fuel pool temperatures within the acceptance limits specified in the Columbia Final Safety Analysis Report [FSAR]. The proposed change to the methodology used to calculate the fuel pool [decay] heat load has no bearing on the probability or consequences of any previously evaluated accident. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The change involves the use of a different methodology for calculating fuel pool decay heat load. This change does not involve any new equipment, it does not change any previously approved acceptance limits, and it does not affect or alter the operation of any equipment. 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 margin of safety provided by the fuel pool cooling system is primarily defined by the difference between the maximum allowed fuel pool temperature and the boiling point of water. The margin of safety is supplemented by the ability to make up water to the spent fuel pool if boiling were to occur. The proposed change in methodology for calculating the fuel pool [decay] heat load does not alter the current temperature limits or acceptance criteria specified in the FSAR and has no effect on the ability to provide make-up water if boiling were to occur. This change will allow Energy Northwest to more accurately calculate the fuel pool [decay] heat load to provide added confidence in the ability of the fuel pool cooling system to accommodate the heat load added to the spent fuel pool during refueling activities. 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: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: April 18, 2006.

Description of amendment request: The proposed change would modify technical specification surveillance requirement 3.6.1.1.2 by changing the test frequency of the drywell-to-suppression chamber bypass leakage test from 24 to 120 months. This proposed amendment also includes testing the suppression chamber-to-drywell vacuum breakers on a 24-month frequency.

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

1. Does the operation of Columbia Generating Station in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes would modify Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.1.2 and add two new SRs, SR 3.6.1.1.3 and SR 3.6.1.1.4. The proposed changes will extend the frequency for the drywell-to-suppression chamber bypass leakage test while maintaining the current leakage testing frequency for the suppression chamber-to-drywell vacuum breakers, and establish leakage acceptance criteria for the suppression chamber-to-drywell vacuum breakers when the valves are tested individually.

The performance of a drywell-to-suppression chamber bypass leakage test or suppression chamber-to-drywell vacuum breaker leakage test is not a precursor to any accident previously evaluated. Thus, the proposed changes to the performance of the leakage tests do not have any affect on the probability of an accident previously evaluated.

The performance of a drywell-to-suppression chamber bypass leakage test or a suppression chamber-to-drywell vacuum breaker leakage test continues to provide assurance that the containment will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not impacted.

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

2. Does the operation of Columbia Generating Station in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes to TS SR 3.6.1.1.2, and the addition of SR 3.6.1.1.3, and SR 3.6.1.1.4 do not affect the assumed performance of any Columbia Generating

Station structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or any new failure mechanisms. This is an administrative change and does not involve the modification, addition or removal of any plant equipment.

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

3. Does the operation of Columbia Generating Station in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The current frequency associated with a drywell-to-suppression chamber bypass leakage test in TS SR 3.6.1.1.2 is 24 months or 12 months if two consecutive tests fail and continues at this frequency until two consecutive tests pass. The proposed change will modify this leakage test frequency to 120 months, or 48 months following one test failure or 24 months if two consecutive tests fail and continues at this frequency until two consecutive tests pass. The proposed change in SR 3.6.1.1.2 frequency is acceptable as the results from previous tests show that the measured drywell-to-suppression chamber bypass leakage at the current TS frequency has been a small percentage of the allowable leakage. Acceptability is further demonstrated by the design requirements applied to the primary containment components and other periodically performed primary containment inspections.

The proposed SR 3.6.1.1.3 will establish a leakage test frequency of 24 months for each suppression chamber-to-drywell vacuum breaker except when the leakage test of SR 3.6.1.1.2 has been performed within the past 24 months. SR 3.6.1.1.3 specifies a leakage limit for each suppression chamber-to-drywell vacuum breaker pathway of less than or equal to 12 percent of the bypass leakage limit of SR 3.6.1.1.2. The proposed SR 3.6.1.1.4 will establish a total leakage limit of less than or equal to 30 percent of the bypass leakage limit of SR 3.6.1.1.2 when the suppression chamber-to-drywell vacuum breakers are tested in accordance with SR 3.6.1.1.3.

TS SR 3.6.1.1.2 drywell-to-suppression chamber bypass leakage test monitors the combined leakage of three types of pathways: (1) The drywell floor and downcomers, (2) piping externally connected to both the drywell and suppression chamber air space, and (3) the suppression chamber-to-drywell vacuum breakers. This amendment would extend the surveillance interval on the passive components of the test (the first two types of pathways), while retaining the current surveillance interval on the active components (suppression chamber-to-drywell vacuum breakers). The proposed changes establish leakage limits for both individual suppression chamber-to-drywell vacuum breakers and the total leakage. Additional testing is required if acceptable results are not achieved.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: David Terao.

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

Date of amendment request: April 12, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specification reactor pressure vessel Pressure and Temperature (P–T) curves.

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

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

Response: No.

The proposed License Amendment (LA) does not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the pressure-temperature curves. The proposed change does not modify the reactor coolant pressure boundary, (i.e., there are no changes in operating pressure, materials, or seismic loading). The proposed change does not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected.

The proposed pressure-temperature curves are generated in accordance with the fracture toughness requirements of 10 CFR 50 Appendix G, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section Xl, Appendix G and Regulatory Guide (R.G.) 1.99, Revision 2[,] ''Radiation Embrittlement of Reactor Vessel Materials.'' A best-estimate calculation of reactor vessel 34 effective full power years (EFPYs) neutron fluence and associated uncertainty has been completed for Pilgrim using the Radiation Analysis Modeling Application (RAMA) methodology. This methodology was previously approved by the NRC. The resulting reactor vessel neutron fluence value was then used in conjunction with R.G. 1.99, [Revision] 2 to determine the adjusted reference temperature (ART) and with ASME Section Xl Appendix G to develop revised P-T curves.

This provides sufficient assurance that the Pilgrim reactor vessel will be operated in a manner that will protect it from brittle fracture under all operating conditions. This proposed license amendment provides compliance with the intent of 10 CFR [Part 50] Appendix G and provides margins of safety that assure reactor vessel 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.

The proposed license amendment does not create the possibility of new or different kind of accident from any accident previously evaluated. The revised pressure-temperature curves are generated in accordance with the fracture toughness requirements of 10 CFR Part 50 Appendix G and ASME Section Xl Appendix G. Compliance with the proposed pressure-temperature curves will ensure the avoidance of conditions in which brittle fracture of primary coolant pressure boundary materials is possible because such compliance with the pressure-temperature curves provides sufficient protection against a non-ductile-type fracture of the reactor pressure vessel. No new modes of operation are introduced by the proposed change. The proposed change will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed change does not affect any activities or equipment and is not assumed in any safety analysis to initiate any accident sequence. This provides sufficient assurance that Pilgrim reactor vessel will be operated in a manner that will protect it from brittle fracture under all operating conditions.

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

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

Response: No.

The proposed license amendment requests the use of revised P-T curves that are based on established NRC and ASME methodologies. A best-estimate calculation of reactor vessel neutron fluence and associated uncertainty has been completed for Pilgrim through 34 EFPY using the NRC approved RAMA methodology. The 34 EFPY reactor vessel neutron fluence value was used in conjunction with R.G. 1.99, [Revision 2] to compute reference temperature shift, and with ASME Section Xl Appendix G to develop revised P-T curves. This provides sufficient margin such that the Pilgrim reactor vessel will be operated in a manner that will protect it from brittle fracture under all operating conditions. Operation within the proposed limits ensures that the reactor vessel materials will continue to behave in a non-brittle manner, thereby preserving the original safety design bases. No plant safetylimits, set points, or design parameters are adversely affected by the proposed changes.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Travis C. McCullough, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601.

Branch Chief: Richard Laufer.

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

Date of amendment request: June 2, 2005.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) reactor coolant system leakage detection instrumentation requirements and actions.

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

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

Response: No. The proposed relocation is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. The associated instrumentation and surveillances are not assumed to be an initiator of any analyzed event, nor are these functions assumed in the mitigation of consequences of accidents. Additionally, the associated required actions for inoperable components do not impact the initiation or mitigation of any accident. 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 involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. The proposed change to relocate current TS requirements to the FSAR [Final Safety Analysis Report], consistent with regulatory guidance and previously approved changes for other stations, are administrative in nature. These changes do not negate any existing requirement, and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications to the FSAR. Additionally, the changes being made to allow additional repair time for inoperable instrumentation will not affect the required leakage limits, which will continue to be monitored at the same required frequency. These compensatory measures, operational limitations, and administrative functions that will be modified are not credited in any design-basis event and do not reflect a margin of safety. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Travis C. McCullough, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601.

Branch Chief: Richard Laufer.

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: November 18, 2005.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The proposed amendment would also include changes to the TS definition of Leakage, TS 3.4.13, ''RCS [Reactor Coolant System] Operational LEAKAGE,'' TS 5.5.9, ''Steam Generator (SG) Program,'' TS 5.6.9, Steam Generator Tube Inspection Report,'' and would add TS 3.4.19, ''Steam Generator (SG) Tube Integrity.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines.''

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 published NSHC determination in its application dated November 18, 2005.

The licensee included a variation from TSTF–449 for Braidwood, Unit 2 and Byron, Unit 2 in that the proposed amendment would also include an effective change to the definition of primary pressure boundary from the hot-leg tube end weld to 17 inches below the top of the hot-leg tube sheet. The proposed amendment would also delete the current TS allowance to use Westinghouse laser welded sleeves as a SG tube repair method. The licensee provided an analyses of the NSHC issue in its application for the plant-specific variations from TSTF–449.

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:

Exelon Generation Company, LLC, (EGC) has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (i.e., 70 FR 10298) as part of the consolidated line item improvement process (CLIIP) item. EGC has concluded that the proposed determination presented in the notice is applicable to Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91 (a), except as discussed below.

The proposed amendment also revises the Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity,'' Revision 4, version of TS 5.5.9, Steam Generator Program, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, ''Provisions for SG tube inspections.'' This proposed license amendment request, in effect, redefines the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tube sheet. This proposed license amendment also deletes the current TS 5.5.9.e.6 and TS 5.5.9.e.10 allowance to use Westinghouse laser welded sleeves as a SG tube repair method.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed TS change by focusing on the three criteria set forth in 10 CFR 50.92 as discussed below:

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

Response: No.

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the SG inspection criteria and delete the allowance to repair SG tubes using Westinghouse laser welded sleeves do not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed changes will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed changes to the SG tube inspection criteria, are the SG tube rupture (SGTR) event and the steam line break (SLB) accident.

During the SGTR event, the required structural integrity margins of the SG tubes will be maintained by the presence of the SG tubesheet. SG tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR [Pressurized Water Reactor] SG Tubes,'' are maintained for both normal and postulated accident conditions.

The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter.

The probability of a SLB is unaffected by the potential failure of a SG tube as this failure is not an initiator for a SLB.

The consequences of a SLB are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the SG creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (*i.e.*, SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (*i.e.*, including indications in the tube end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by up to approximately a factor of three. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.104 gpm [gallons per minute] (150 gpd [gallons per day]) per TS 3.4.13, ''RCS Operational Leakage,'' the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by approximately 0.2 gpm. This value is well within the assumed accident leakage rate of 0.5 gpm discussed in Updated Final Safety Analysis Table 15.1–3, ''Parameters Used in Steam Line Break Analyses.'' Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet. Therefore, the consequences of a SLB accident remain unaffected.

Based on the above discussion, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

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

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

Response: No.

The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' Revision 1 and Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' are used as the bases in the development of the limited hot leg tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, ''Reactor coolant pressure boundary,'' GDC 15, ''Reactor coolant system design,'' GDC 31, ''Fracture prevention of reactor coolant pressure boundary,'' and GDC 32, ''Inspection of reactor coolant pressure boundary,'' by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse letter LTR–CDME–05–32, ''Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron Unit 2 and Braidwood Unit 2,'' Revision 2, dated August 2005, defines a length of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot leg tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited hot leg tubesheet inspection depth criteria.

Therefore, the proposed changes do not involve a significant hazards consideration under the criteria set forth in 10 CFR 50.92(c).

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. Brad J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: January 25, 2006.

Description of amendment request: The proposed amendment would revise the Updated Final Safety Analysis Report (UFSAR) to allow the use of automatic load tap changers (LTCs) to operate in automatic mode on the reserve auxiliary transformers (RATs) to compensate for potential offsite power voltage fluctuations, in order to ensure that acceptable voltage is maintained for safety related equipment.

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 requested change allows the automatic operation mode of the LTC. The only accident previously evaluated for which the probability is potentially affected by the change is the loss of offsite power (LOOP). A failure of the LTC while in automatic operation mode that results in decreased voltage to the ESS [essential service system] buses could cause a LOOP. This could occur in two ways. A failure of the LTC controller that results in rapidly decreasing the voltage to the emergency buses is the most severe failure mode. However, a backup controller is provided with the LTC that makes this failure unlikely. A failure of the LTC controller to respond to decreasing grid voltage is less severe, since grid voltage changes occur slowly. In both of the above potential failure modes, operators will take manual control of the LTC to mitigate the effects of the failure. Thus, the probability of a LOOP is not significantly increased.

The proposed change has no effect on the consequences of a LOOP, since the emergency diesel generators provide power to safety related equipment following a LOOP. The emergency diesel generators are not affected by the proposed change.

The probability of other accidents previously evaluated is not affected, since the proposed change does not affect the way plant equipment is operated and thus does not contribute to the initiation of any of the previously evaluated accidents.

The LTC is equipped with a backup controller, which controls the LTC in the event of primary controller failure. Additionally, operator action is available to prevent a sustained high voltage condition from occurring. Damage due to over-voltage is time-dependent. Therefore, damage of safety related equipment is extremely unlikely, and the consequences of these accidents are not significantly increased. The only way in which the consequences of other previously evaluated accidents could be affected is if a failure of the LTC, while in automatic operation mode, led to a sustained high voltage condition, which resulted in damage to safety related equipment that is used to mitigate an accident.

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 involves functions that provide offsite power to safety related equipment for accident mitigation. Thus, the proposed change potentially affects the consequences of previously evaluated accidents (as addressed in Question 1), but does not result in any new mechanisms that could initiate damage to the reactor and its principal safety barriers (i.e., fuel cladding, reactor coolant system, or primary containment).

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

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

Response: No.

The proposed change does not affect the inputs or assumptions of any of the analyses that demonstrate the integrity of the fuel cladding, reactor coolant system, or containment during accident conditions. The allowable values for the degraded voltage protection function are unchanged and will continue to ensure that the degraded voltage protection function actuates when required, but does not actuate prematurely to cause a LOOP. Automatic operation of the LTC increases margin by reducing the potential for transferring to the EDGs [emergency diesel generators] during an event.

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 Exelong Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: February 10, 2006.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.3.5.1, ''Emergency Core Cooling System (ECCS) Instrumentation,'' to correct a Perry Nuclear Power Plant (PNPP)-specific issue and establish consistency with the improved standard technical specifications (ISTS). Specifically, Sub-actions B.1.2.1 and B.1.2.2, which were added into PNPP TS 3.3.5.1 during the ISTS conversion process, will be deleted. PNPP Required Action B.1 will then match the ISTS Required Action B.1. As a result, actions with a 1-hour completion time will only be required for the annulus exhaust gas treatment (AEGT) system if a loss of initiation capability in both divisions actually exists for an AEGT initiation function, as originally intended.

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

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

There are no physical modifications being made to any plant system or component. The only change is to a Required Action within the Technical Specifications. The revised Technical Specification requirements do not impact initiators of previously evaluated accidents or transients.

The specification being revised is associated with a system used to mitigate the consequences of accidents. The change does not affect how the AEGT system is controlled, operated, or tested. The intent of Required Action B.1 for the ECCS Instrumentation, specifically, a loss of initiation capability check, is maintained by the changes being proposed. The wording of Required Action B.1 ensures appropriate actions are taken when a loss of initiation capability exists, by declaring the supported systems inoperable. This action is consistent with the current requirements.

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

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

There are no physical modifications being made to any plant system or component, and the proposed change introduces no new method of operation for the plant, or its systems or components. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The change to the ECCS Instrumentation Required Action continues to ensure that a check is performed to determine if one or more of the ECCS Instrumentation Functions has lost its capability to actuate the Division 1 and 2 low-pressure ECCS, the AEGT subsystems, and the associated diesel generators. It continues to direct appropriate actions if such a loss of initiation capability is found. Therefore, the necessary function of the Technical Specification requirements is maintained, and 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: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: February 16, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator (SG) tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The availability of this TS improvement was announced in the **Federal Register** on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on March 2, 2005 (70 FR 10298) as part of the CLIIP. The licensee affirmed the applicability of the model NSHC determination in its application dated February 16, 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:

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 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 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 to the TS 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. 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 and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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.

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

Date of amendment request: July 9, 2004 (TS–436).

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement 3.6.1.3.10 to increase the allowed main steam isolation valve (MSIV) leak rate from 11.5 standard cubic feet per hour (scfh) per valve, to 100 scfh for individual MSIVs with a 150 scfh combined leakage for all four main steam lines.

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.

TVA proposes to utilize the main steam drain lines to preferentially direct MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this approach, the main steam lines, steam drain piping, and the main condenser are used to mitigate the consequences of an accident to limit potential doses below the limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

Seismic verification walkdowns and evaluations of bounding piping/supports were performed to demonstrate the main steam line piping and components that comprise the Alternate Leakage Treatment (ALT) path were rugged and able to perform the safety function of MSIV leakage control following a Design Basis Earthquake (DBE). Thus, it has been concluded the components in the MSIV alternate leakage treatment flow path can be relied upon to maintain structural integrity.

Therefore, the proposed amendment does not involve changes to structures, components, or systems which would affect the probability of an accident previously evaluated in the Browns Ferry Updated Final Safety Analysis Report (UFSAR).

A plant-specific radiological analysis has been performed to assess the effects of the proposed increase in MSIV leakage acceptance criteria in terms of off-site doses and main control room dose. The analysis shows the dose contribution from the proposed increase in leakage acceptance criteria is acceptable compared to doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

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

Response: No.

The proposed changes require the use of the main steam piping and the condenser to process MSIV leakage. This additional function does not compromise the reliability of these systems. They will continue to function as intended and not be subject to a failure of a different kind than previously considered. In addition, MSIV functionality will not be adversely impacted by the increased leakage limit. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change to Surveillance Requirement 3.6.1.3.10, to increase the allowable MSIV leakage, does not involve a significant reduction in the margin of safety. The allowable leak rate specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass containment. The results of the re-analysis supporting these changes were evaluated against the dose limits contained in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel. Sufficient margin relative to the regulatory limits is maintained even when conservative assumptions and methods are utilized. 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: 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 Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: August 16, 2004 (TS—447).

Description of amendment request: The proposed amendment would extend the channel calibration frequency requirements for instrumentation in the high pressure coolant injection, reactor core isolation cooling, and reactor water core isolation cooling systems.

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

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

Response: No.

The proposed changes extend the channel calibration surveillance frequency of instrumentation used for the high area temperature isolation of the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), and the reactor water clean-up (RWCU) systems. The allowable trip point value for three sets of RCIC instruments on each unit and for two sets of RWCU instruments on Unit 1 are also revised. The calibration surveillance frequency is extended to 24 months from 92 days (for the HPCI and RCIC high area temperature instrumentation) and from 122 days (for the RWCU high area temperature instrumentation). Under certain circumstances, Technical Specifications (TS) SR [Surveillance Requirement] 3.0.2 would allow a maximum surveillance interval of 30 months for an SR having a nominal 24-month performance frequency. Instrumentation scaling and setpoint calculations performed in accordance with the guidelines of Generic Letter 91–04 have shown that the reliability of the affected protection instrumentation will be preserved for the maximum allowable calibration surveillance interval. The Unit 1 instrumentation will be physically modified to be essentially identical to that installed on Unit 2 and Unit 3 prior to restart of Unit 1. 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 changes extend the channel calibration surveillance frequency of instrumentation used for the high area temperature isolation of the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), and the reactor water clean-up (RWCU) systems. The allowable trip point value for three sets of RCIC instruments on each unit and for two sets of RWCU instruments on Unit 1 are also revised. The instrumentation will function in the same way following the amendment as it functions currently. Hence, the changes do not create the possibility of any new failure mechanisms. Note that the Unit 1 instrumentation will be modified to be essentially identical to that installed on Unit 2 and Unit 3 prior to restart of Unit 1. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed changes extend the channel calibration surveillance frequency of instrumentation used for the high area temperature isolation of the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), and the reactor water clean-up (RWCU) systems. The allowable trip point value for three sets of RCIC instruments on each unit and for two sets of RWCU instruments on Unit 1 are also revised. Instrumentation scaling and setpoint calculations performed in accordance with the guidelines of Generic Letter 91–04 have shown safety margins are preserved with the extended surveillance frequency and the revised TS allowable values. 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: 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 (TVA), Docket No. 50–390, Watts Bar Nuclear Plant (WBN), Unit 1, Rhea County, Tennessee

Date of amendment request: May 8, 2006 (TS–06–09).

Description of amendment request: The proposed amendment would revise the limiting condition for operation for Technical Specification (TS) Section 3.7.9, ''Ultimate Heat Sink.'' The maximum essential raw cooling water (ERCW) temperature limit associated with Surveillance Requirement 3.7.9.1 would increase from 85 degrees Fahrenheit (°F) to 88 °F. This proposed change is based on evaluations of the ERCW system and the ultimate heat sink (UHS) functions and maximum temperatures that will satisfy the associated safety functions.

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

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

Response: No.

The proposed change to increase the UHS maximum temperature will not adversely alter the function, design, or operating practices for plant systems or components. The UHS is utilized to remove heat loads from plant systems during normal and accident conditions. This function is not expected or postulated to result in the generation of any accident and continues to adequately satisfy the associated safety functions with the proposed changes. Therefore, the probability of an accident presently evaluated in the safety analyses will not be increased. The heat loads, that the UHS is designed to accommodate, have been evaluated with the higher temperature limit. The result of these evaluations is that there is existing margin associated with the systems that utilize the UHS for normal and accident conditions. These margins are sufficient to accommodate the postulated normal and accident heat loads with the proposed changes to the UHS. Since the safety functions of the UHS are maintained, the systems that ensure acceptable offsite dose consequences will continue to operate as designed. The change in the maximum calculated containment pressure associated with the design basis loss-of-coolant-accident (LOCA) remains below the American Society of Mechanical Engineers (ASME) Code design internal pressure. Therefore, the consequence of any accident will be the same as those previously analyzed.

Since the UHS safety function will continue to meet accident mitigation requirements and limit dose consequences to acceptable levels, TVA has concluded that the proposed TS 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 UHS function provides accident mitigation capabilities and serves as a heat sink for normal and upset plant conditions; the UHS is not an initiator of any accident. By allowing the proposed change in the UHS temperature requirements, only the parameters for UHS operation are changed while the safety functions of the UHS and systems that transfer the heat sink capability continue to be maintained. The proposed change does not impact the response of the systems and components assumed in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident evaluated.

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

Response: No.

The proposed change has been evaluated for systems that are needed to support accident mitigation functions as well as normal operational evolutions. Operational margins were found to exist in the systems that utilize the UHS capabilities such that these proposed changes will not result in the loss of any safety function necessary for normal or accident conditions. The ERCW system has excess flow capacity that will accommodate the increased flows necessary for the proposed temperature increase. While operating margins have been reduced by the proposed changes, safety margins have been maintained as assumed in the accident analyses for postulated events. The proposed change results in an increase in the maximum calculated containment peak pressure. However, the change in the maximum calculated containment peak pressure associated with the design basis LOCA is a small percentage of the margin between the current maximum calculated containment peak pressure and the ASME Code design internal pressure. This aspect of the proposed change does not involve a significant reduction in a margin of safety. Additionally, the proposed changes do not require any further modification of component setpoints or operating provisions that are necessary to maintain margins of safety established by the WBN design (the shutdown board room chillers were physically modified to operate properly at the 88 degree F UHS temperature). 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: April 14, 2005, as supplemented by letter dated December 21, 2005.

Description of amendment request: The amendment would revise the Technical Specifications (TSs) by (1) adding a new TS 3.1.9, ''RCS [Reactor Coolant System] Boron Limitations <500 °F,'' and (2) revising TS 3.3.1, ''Reactor Trip System (RTS) Instrumentation.''

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no design changes. The design of the reactor trip system (RTS) instrumentation and engineered safety feature actuation system (ESFAS) instrumentation will be unaffected and these protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to this amendment request will be maintained.

The proposed changes will not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained other than extending the OPERABILITY requirements for RTS trip Function 2.b (Power Range Neutron Flux—Low) to the upper portion of MODE 3. The proposed changes will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits.

As discussed previously [in the application,] the proposed change[s] will add more restrictive requirements in the form of a new LCO [limiting condition for operation] 3.1.9 and an expanded LCO Applicability for RTS trip Function 2.b, Power Range Neutron Flux—Low, to provide mitigative capability in the event of an uncontrolled RCCA [rod cluster control assembly] bank withdrawal event postulated to occur during low power or subcritical (startup) conditions.

There will be no change[s] to normal plant operating parameters or accident mitigation performance. None of the proposed changes will initiate any accidents; therefore, the probability of an accident will not be increased. There will be no degradation in the performance of, nor an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation.

All accident analysis acceptance criteria will continue to be met with the proposed changes. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [Final Safety Analysis Report for Callaway]. The applicable radiological dose acceptance criteria will continue to be met.

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

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

Response: No.

There are no proposed design changes nor are there any changes in the method by which any safety-related plant SSC performs its safety function. [These changes] will not affect the normal method of plant operation or change any operating parameters. No equipment performance requirements will be affected other than the more restrictive Applicability requirements being imposed on RTS trip Function 2.b, Power Range Neutron Flux—Low, in the upper portion of MODE 3. The proposed changes will not alter any assumptions made in the safety analyses.

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

The proposed amendment will not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

Response: No.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The applicable radiological dose consequence acceptance criteria will continue to be met.

The proposed changes do not eliminate any RTS or ESFAS surveillances or alter the Frequency of surveillances required by the Technical Specifications. More restrictive changes are proposed by virtue of a new LCO 3.1.9 on [RCS] boron requirements when the RCS temperature is below 500 °F and by virtue of extending the Applicability of RTS trip Function 2.b, Power Range Neutron Flux—Low, to the upper portion of MODE 3. The nominal RTS and ESFAS trip setpoints will remain unchanged. None of the acceptance criteria for any accident analysis will be changed.

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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Branch Chief: David Terao.

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

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

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

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

Date of amendment request: April 20, 2006.

Brief description of amendment request: The proposed amendments would reinstate the previous reactor coolant system pressure and temperature limits, low temperature overpressure protection system (LTOPS) setpoint, and (LTOPS) enable temperature basis that were approved by the NRC staff on December 28, 1995, as License Amendments Nos. 207 and 207 for Surry 1 and 2.

*Date of publication of individual notice in **Federal Register:*** April 28, 2006 (71 FR 25249)

Expiration date of individual notice: 30 day expiration date, May 30, 2006, and 60 day expiration date, June 27, 2006.

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: June 20, 2005.

Brief Description of amendments: The amendments revise the Technical Specification (TS) Surveillance Requirement 3.6.1.6.2 of 3.6.1.6, ''Suppression Chamber-to-Drywell Vacuum Breakers'' for the frequency of functionally testing the suppression chamber-to-drywell vacuum breakers.

Date of issuance: May 5, 2006.

Effective date: May 5, 2006.

Amendment Nos.: 240 and 268.

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

Date of initial notice in ***Federal Register:*** August 16, 2005 (70 FR 48202).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 5, 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: August 18, 2005, as supplemented by letter dated February 15, 2006.

Brief description of amendment: This amendment authorizes the use of fire-resistive electrical cables in lieu of the alternatives specified in Section C5.b.2 of Branch Technical Position Chemical Engineering Branch 9.5–1 (NUREG–0800), '' Guidelines for Fire Protection for Nuclear Power Plants,'' dated July 1981, for Fire Areas 12–A–CR, 1–A–CSRA, 1–A–CSRB, 1–A–SWGRA, 1–A–SWGRB, and 1–A–BAL–B.

Date of issuance: May 1, 2006.

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

Amendment No. 123.

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

Date of initial notice in ***Federal Register:*** November 8, 2005.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 1, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 14, 2005, as supplemented January 11, 2006.

Brief description of amendment: The proposed change modifies the Millstone Power Station, Unit No. 2 reactor coolant system heatup and cooldown limits Technical Specification (TS) 3.4.9.1, ''Reactor Coolant System''. The associated TS bases will be updated to address the proposed change.

Date of issuance: May 3, 2006.

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

Amendment No.: 292.

Facility Operating License No. DPR–65: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** August 30, 2005 (70 FR 51379). The supplement dated January 11, 2006, provided clarifying information that did not change the scope of the proposed amendment as described in the original 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 May 3, 2006.

No significant hazards consideration comments received: No.

Duke Power Company, LLC Docket No. 50–287, Oconee Nuclear Station, Unit 3, Oconee County, South Carolina

Date of application of amendment: August 18, 2005, supplemented September 15, 2005, and January 5 and April 6, 2006.

Brief description of amendment: The amendment revised Technical Specifications 3.5.2.6 and 3.5.3.6 to accommodate the replacement of the reactor building emergency sump suction inlet trash racks and screens with strainers. Similar amendments were issued for Units 1 and 2 on November 1, 2005; however, the amendment for Unit 3 was not issued at that time since the licensee had not completed its evaluation of the impact of pipe whip, jet impingement and internally generated missiles for Unit 3.

Date of Issuance: May 4, 2006.

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

Amendment No.: 350.

Renewed Facility Operating License No. DPR–55: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 31, 2005 (70 FR 51852).

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the initial **Federal**

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 24, 2004.

Brief description of amendments: These amendments implement 25 generic Technical Specification (TS) changes previously approved by the NRC staff as part of the Technical Specifications Task Force (TSTF). The TSTF change travelers and proposed changes are:

1. TSTF–5, an administrative change to TS 2.2 to remove reporting requirements that are already in the regulations 10 CFR, Sections 50.36 and 50.73;
2. TSTF–208, an extension of the time allowed to reach MODE 2 once a TS 3.0.3 condition is identified, from the current 7 hours to 10 hours;
3. TSTFs–222 and 229, changes to TS 3.1.4 to allow scram time testing on only affected rods when an outage is short and only a limited number of fuel assemblies are moved and to require the Minimum Critical Power Ratio to be determined after scram time testing;
4. TSTFs–297 and 227, changes to TSs 3.3.2.2, 3.3.4.1, and 3.3.4.2 to allow reactor feedwater pumps and main turbine valves to be removed from service if their trip function is compromised;
5. TSTF–295, a clarification in Table 3.3.3.1–1 that penetration flow paths, not just valve positions, are to be considered;
6. TSTF–275, a clarification Table 3.3.5.1–1 that certain emergency core cooling system (ECCS) instrumentation needs to be operable when ECCS and ECCS support systems are required to be operable;
7. TSTF–306, changes to TS 3.3.6.1 to allow penetration flow paths to be opened intermittently under administrative controls and to set apart the Traversing In-core Probe system isolation as a separate function;
8. TSTF–416, changes to TSs 3.5.1 and 3.5.2 to allow the low pressure coolant injection subsystems to be considered operable during alignment and operation in the decay heat removal mode;
9. TSTF–17, a change to TS 3.6.1.2 to extend the containment air lock interlock mechanism testing frequency from 6 months to 2 years to coincide with refueling outage frequency;
10. TSTFs–30, 323, 45, 46, and 269, changes to TSs 3.6.1.3 and 3.6.4.2 related to primary and secondary containment isolation valve completion times, isolation times, and status verification;
11. TSTF–322, a clarification in TS 3.6.4.1 of the intent of secondary containment drawdown tests;
12. TSTF–276, Revision 2, a change to TS 3.8.1 to allow certain emergency diesel generator (EDG) testing to continue even if the stated power factor cannot be attained;
13. TSTF–404, a change to TS 3.1.8 to revise required actions when one valve is inoperable in one or more scram discharge volume vent and drain lines, as part of the consolidated line item improvement process;
14. TSTF–65 Revision 1, a change to allow the use of generic organizational titles in the TSs, as opposed to plant-specific titles;
15. TSTF–299, a clarification in TS 5.2.2 of the intent of refueling cycle intervals with respect to system leak test requirements;
16. TSTF–279, a deletion in TS 5.5.6 of the reference to ''applicable supports'' as part of the description of the Inservice Testing Program;
17. TSTF–118, a change to TS 5.5.9 to apply the provisions of Surveillance Requirement (SR) 3.0.2 (25% extension interval) and SR 3.0.3 (missed surveillance actions) to EDG fuel oil testing surveillances;
18. TSTF–106, Revision 1, a clarification in TS 5.5.9 that the American Society for Testing and Materials standard for EDG fuel oil applies only to new fuel being received; and
19. TSTF–152, a change to the Radioactive Effluent Release Report to ensure that a common report for both units combines sections common to both units.

Date of issuance: May 10, 2006.

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

Amendments Nos.: 259 and 262.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the TSs.

Date of initial notice in ***Federal Register:*** September 28, 2004 (69 FR 57985) and October 26, 2004 (69 FR 62476).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: December 19, 2006.

Description of amendment request: The amendment deletes Technical Specification (TS) 6.8.1.2a, ''Occupational Radiation Exposure Report [ORER],'' TS 6.8.1.2.c, regarding challenges to pressurizer relief and safety valves and TS 6.8.1.5, ''Monthly Operating Report [MOR],'' as described in the Notice of Availability published in the **Federal Register** on June 23, 2004 (69 FR 35067).

Date of issuance: May 5, 2006.

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

Amendment No.: 109.

Facility Operating License No. NPF–86: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** February 14, 2006 (71 FR 7808).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 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: October 12, 2005.

Brief description of amendment: The amendment revised Technical Specification (TS) Section 3.4.9, ''RCS [reactor coolant system] Pressure and Temperature (P/T) Limits,'' curves 3.4.9–1, ''Pressure/Temperature Limits for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown,'' 3.4.9–2, ''Pressure/Temperature Limits for Inservice Hydrostatic and Inservice Leakage Tests, and 3.4.9–3, ''Pressure/Temperature Limits for Criticality,'' to remove the cycle operating restriction and replace it with a limitation of 30 effective full-power years (EFPY).

Date of issuance: April 27, 2006.

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

Amendment No.: 219.

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

Date of initial notice in ***Federal Register:*** January 3, 2006 (71 FR 150).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 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: December 30, 2005.

Brief description of amendment: The amendment established a combined leakage rate limit for the sum of the four main steam line leakage rates that is equal to four times the current individual main steam isolation valve leakage rate limit.

Date of issuance: May 2, 2006.

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

Amendment No.: 220.

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

Date of initial notice in ***Federal Register:*** February 28, 2006 (71 FR 10073)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 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: January 30, 2006.

Brief description of amendment: The amendment 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 consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.8 is added to the TS to provide this allowance and define the requirements and limitations for its use.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, 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 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated January 30, 2006.

Date of issuance: May 2, 2006.

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

Amendment No.: 221.

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

Date of initial notice in ***Federal Register:*** February 28, 2006 (71 FR 10074).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 29, 2004, as supplemented on November 23, 2004; January 20, February 28, April 12, 2005; and March 10, 2006.

Brief description of amendment: The amendment revised the MNGP licensing basis by selectively implementing the alternative source term for the postulated fuel handling accident, leading to revision of portions of the Technical Specifications to reflect this change in licensing basis.

Date of issuance: April 24, 2006.

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

Amendment No.: 145.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 18, 2005 (70 FR 2891)

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 24, 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: August 11, 2005.

Brief description of amendment: The 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 consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.8 is added to the TS to provide this allowance and define the requirements and limitations for its use.

Date of issuance: March 1, 2006.

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

Amendment No.: 238.

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

Date of initial notice in ***Federal Register:*** December 6, 2005 (70 FR 72674)

The Commission's related evaluation of the amendment is contained in a safety evaluation dated March 1, 2006.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: October 19, 2005.

Brief description of amendments: The 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 consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.8 is added to the TS to provide this allowance and define the requirements and limitations for its use.

Date of issuance: March 7, 2006.

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

Amendment Nos.: Unit 1—185; Unit 2—187

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 20, 2005 (70 FR 75495).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 9, 2004, as supplemented by letters dated November 18 and December 5, 2005.

Brief description of amendment: The amendment authorizes modification to the Updated Final Safety Analysis Report (UFSAR) to include a revision to the methodology for splicing reinforcing steel bars during restoration of the Unit 1 concrete shield building dome as part of the steam generator replacement project.

Date of issuance: April 27, 2006.

Effective date: As of the date of issuance and shall be implemented as part of the next UFSAR update made in accordance with 10 CFR 50.71(e).

Amendment No. 60.

Facility Operating License No. NPF–90: Amendment authorizes revision of the Updated Final Safety Analysis Report.

*Date of initial notice in the **Federal Register:*** January 4, 2005 (70 FR 405). 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 April 27, 2006.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: October 30, 2003, as supplemented by letters dated August 31 and November 18, 2005, and March 6, 2006.

Brief description of amendment: The amendment increases the completion times (CTs) for Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' and adds requirements on the diesel generators at the Sharpe Station when a diesel generator at Wolf Creek Generating Station is in an extended CT greater than 72 hours. The proposed changes to TS 3.8.9, ''Distribution Systems—Operating,'' are withdrawn. The amendment also revises a page in the license and adds conditions to Appendix D, ''Additional Conditions,'' of the license.

Date of issuance: April 26, 2006.

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

Amendment No.: 163.

Facility Operating License No. NPF–42. The amendment revised the license including Appendix D, ''Additional Conditions,'' and Appendix A, ''Technical Specifications.''

*Date of initial notice in **Federal Register:*** January 6, 2004 (69 FR 700). The supplemental letters dated August 31 and November 18, 2005, and March 2, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 26, 2006.

No significant hazards consideration comments received: No

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: November 3, 2005, and supplemental letters dated February 21 and March 28, 2006.

Brief description of amendment: The amendment revised the Technical Specifications associated with steam generator tube integrity consistent with Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on May 6, 2005 (70 FR 24126).

Date of issuance: May 8, 2006.

Effective date: The license amendment is effective as of its date of issuance and shall be implemented prior to the entry into Mode 5 in the restart from Refueling Outage 15, which is scheduled to begin in October 2006.

Amendment No.: 164.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 6, 2005 (70 FR 72676) The supplemental letters dated February 21 and March 28, 2006, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 15th day of May 2006.

For the Nuclear Regulatory Commission

Catherine Haney,

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

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OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Trade Policy Staff Committee; Initiation of Environmental Review of Proposed Free Trade Agreement Between the United States and Malaysia; Public Comments on Scope of Environmental Review

AGENCY: Office of the United States Trade Representative.

ACTION: Notice and request for comments.

SUMMARY: This publication gives notice that, pursuant to the Trade Act of 2002, and consistent with Executive Order 13141 (64 FR 63169) (Nov. 18, 1999) and its implementing guidelines (65 FR 79442), the Office of the United States Trade Representative (USTR), through the Trade Policy Staff Committee (TPSC), is initiating an environmental review of the proposed free trade agreement (FTA) between the United States and Malaysia. The TPSC is requesting written comments from the public on what should be included in the scope of the environmental review, including the potential environmental effects that might flow from the free trade agreement and the potential implications for U.S. environmental laws and regulations, and identification of complementarities between trade and environmental objectives such as the promotion of sustainable development. The TPSC also welcomes public views on appropriate methodologies and sources of data for conducting the review. Persons submitting written comments should provide as much detail as possible on the degree to which the subject matter they propose for inclusion in the review may raise significant environmental issues in the context of the negotiation.

DATES: Public comments should be received no later than July 7, 2006.

ADDRESSES:

Submissions by electronic mail: FR06017@ustr.eop.gov.

Submissions by facsimile: Gloria Blue, Executive Secretary, Trade Policy Staff Committee, at (202) 395–6143.

FOR FURTHER INFORMATION CONTACT: For procedural questions concerning public comments, contact Gloria Blue,