request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (71 FR 37614).

For further details with respect to the action, see (1) the application for amendment dated July 7, 2005, as supplemented by letters dated August 15, September 30, and December 6, 9, and 22, 2005, and January 11 and 25, February 16, March 3 and 24, and May 9 and 19, 2006, (2) Amendment No. 97 to License No. DPR–18, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, Public File Area O1 F21,11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically 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.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC Public Document Room Reference staff by telephone at 1–800–397–4209, or 301–415–4737, or by e-mail to *pdr@nrc.gov.*

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

For the Nuclear Regulatory Commission.

Patrick D. Milano,

Senior Project Manager,Plant Licensing Branch I–1, Division of Operating Reactor Licensing,Office of Nuclear Reactor Regulation.

[FR Doc. E6–11320 Filed 7–17–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act; Notice of Meeting

DATES: Weeks of July 17, 24, 31, August 7, 14, 21, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of July 17, 2006

There are no meetings scheduled for the Week of July 17, 2006.

Week of July 24, 2006—Tentative

Wednesday, July 26, 2006

1:50 p.m. Affirmation Session (Public Meeting) (Tentative).

a. *Pa'ina Hawaii, LLC,* unpublished April 27, 2006 Memorandum and Order (accepting the intervenor's and NRC Staff's Joint Stipulation regarding two admitted environmental contentions) (Tentative).

b. *David Geisen,* LBP–06–13 (May 19, 2006) (Tentative).

c. *Exelon Generation Company, LLC* (Early Site Permit for Clinton ESP), *System Energy Resources, Inc.* (Early Site Permit for Grand Gulf ESP) (Tentative).

d. *Florida Power & Light Co., et al.,* Docket Nos. 50–250–LT, *et al.,* International Brotherhood of Electrical Workers' ''Petition to File Motion to Intervene and Protest Out-of-Time'' and ''Motion for Hearing and Right to Intervene and Protest'' (Tentative).

Thursday, July 27, 2006

9:30 a.m. Briefing on Office of International Programs (OIP) Programs, Performance, and Plans (Public Meeting) (Contact: Karen Henderson, 301–415–0202).

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

1:30 p.m. Briefing on Equal Employment Opportunity (EEO) Programs. (Public Meeting) (Contact: Barbara Williams, 301–415–7388).

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

Week of July 31, 2006—Tentative

There are no meetings scheduled for the Week of July 31, 2006.

Week of August 7, 2006—Tentative

There are no meetings scheduled for the Week of August 7, 2006.

Week of August 14, 2006—Tentative

There are no meetings scheduled for the Week of August 14, 2006.

Week of August 21, 2006—Tentative

There are no meetings scheduled for the Week of August 21, 2006.

* * * * *

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

* * * * *

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

* * * * *

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g. braille, large print), please notify the NRC's Disability Program Coordinator, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by e-mail at *DLC@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

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

Dated: July 13, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–6302 Filed 7–14–06; 9:59 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 23, 2006 to July 6, 2006. The last biweekly notice was published on July 5, 2006 (71 FR 38180).

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.*

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

Date of amendment request: May 15, 2006.

Description of amendment request: The amendment would revise the Technical Specification (TS) requirements related to steam generator tube integrity. The proposed changes are generally consistent with 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 proposed amendment includes changes to licensing pages to delete License Condition 2.c.(8), ''Repaired Steam Generators;'' changes to TS 3.1.6, ''LEAKAGE;'' changes to TS Section 3.1.1.2, ''Steam Generators and Steam Generator (SG) Tube Integrity;'' revising TS Section 4.19, ''Steam Generator (SG) Tube Integrity;'' adding new TS 6.9.6, ''Steam Generator Tube Inspection Report;'' and adding new TS 6.19, ''Steam Generator (SG) Program.''

Basis for proposed no significant hazards consideration determination (NSHC): The NRC staff published a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model NSHC determination, using the CLIIP. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated May 15, 2006. As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

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

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

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

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

The SG performance criteria proposed change to the TSs identifies the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design-basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TSs. The program, defined by NEI [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 the primary-to-secondary leak rate after the accident is 1 gallon per minute with no more than 500 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

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

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

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

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

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

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

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

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

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

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

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: June 1, 2006.

Description of amendment request: The proposed amendments would revise the Updated Final Safety Analysis Report (UFSAR) to incorporate the use of a fiber-reinforced polymer (FRP) system to strengthen existing masonry walls against tornado effects.

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

Response: Physical protection from a tornado event is a design basis criterion rather than a requirement of a previously analyzed UFSAR accident analysis.

The current licensing basis (CLB) for Oconee states that systems, structures, and components (SSC's) required to shut down and maintain the units in a shutdown condition will not fail as a result of damage caused by natural phenomena.

The in-fill masonry walls to be strengthened using an FRP system are passive, non-structural elements. The use of an FRP system on existing Auxiliary Building masonry walls will allow them to resist uniform pressure loads resulting from a tornado and will not adversely affect the structure's ability to withstand other design basis events such as earthquakes or fires. Therefore, the proposed use of FRP on existing masonry walls will not significantly increase the probability or consequences of an accident previously evaluated.

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

Response: The final state of the FRP system is passive in nature and will not initiate or cause an accident. More generally, this understanding supports the conclusion that the potential for new or different kinds of accidents is not created.

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

Response: The application of an FRP system to existing auxiliary building masonry walls will either act to restore the margin of safety described in the UFSAR, *e.g.*, the Unit 3 Control Room north wall, or enhance the margin of safety, *e.g.*, the West Penetration Room walls, by increasing the walls' ability to resist tornado-induced differential pressure and/or tornado wind. Consequently, 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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: May 22, 2006.

Description of amendment request: The proposed license amendment request would revise: (1) Surveillance Requirement (SR) 3.8.1.11 to remove the MODE restriction from Note 2 for Diesel Generator (DG)–3 only, (2) SR 3.8.1.12 to remove the MODE restriction from Note 2 for DG–3 only, (3) SR 3.8.1.16 to remove the MODE restriction from the Note for DG–3 only, and (4) Revise SR 3.8.1.19 to remove the MODE restriction from Note 2 for DG–3 only.

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 DG and its associated emergency loads are accident mitigating features, not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of the requested amendment. The design of plant equipment is not being modified by these proposed changes. The capability of DG–1 and DG–2 to supply power to their safety related buses as designed will not be compromised by permitting performance of DG–3 testing during power operations. Columbia's Technical Specifications require the RCIC [reactor core isolation cooling] system to be operable whenever this testing is performed at power. This ensures that the high-pressure injection function is maintained during the time the HPCS injection valve is disabled during testing. In the event of a design basis accident during testing, the HPCS [high-pressure core spray] system could be returned to service well within the 14-day outage time allowed by Technical Specifications. Additionally, the ability of the Standby Liquid Coolant (SLC) system to perform its design safety function would not be affected because SLC is connected downstream of the HPCS injection valve. Therefore, there would be no significant impact on any accident consequences.

Based on the above, the proposed change to permit certain DG surveillance tests to be performed during plant operation will have no effect on accident probabilities or consequences. Therefore, the proposed change does 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.

No new accident causal mechanisms would be introduced as a result of NRC approval of this amendment request since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration with the exception of the plant mode in which the testing is conducted. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

Based on the above, implementation of the proposed changes would not create the possibility of a new or different kind of accident from any accident 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.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the DG do not affect the operability requirements for the DG, as verification of such operability will continue to be performed as required. Continued verification of operability supports the capability of the DG to perform its required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment. In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this, and the above basis, no safety margins will be impacted.

Energy Northwest concludes that there is no significant reduction in the margin of safety.

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

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

NRC Branch Chief: David Terao.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: April 24, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) consistent with the NRC-approved Revision 4 to TS Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated April 24, 2006. Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

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

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

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

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

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

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

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

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

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

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

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

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes. Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

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

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

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

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: May 25, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) consistent with the NRC-approved Revision 4 to TS Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated May 25, 2006.

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

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

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

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

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

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

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

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

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

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

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

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

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes. Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

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

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

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

NRC Branch Chief: Michael L. Marshall, Jr.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: April 27, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) consistent with the NRC-approved Revision 4 to TS Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated April 27, 2006.

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

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

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

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

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

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

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

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

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

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

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

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

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes. Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

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

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

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

NRC Branch Chief: Michael L. Marshall, Jr.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: January 18, 2006.

Description of amendment request: The proposed amendment would delete the reference to the hydrogen monitors in Technical Specification (TS) 3.6.11, ''Accident Monitoring Instrumentation'' consistent with the NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.''

The NRC staff issued a notice of availability of ''Model Application Concerning Technical Specification Improvement To Eliminate Hydrogen Recombiner Requirement, and Relax the Hydrogen and Oxygen Monitor Requirements for Light Water Reactors Using the Consolidated Line Item Improvement Process (CLIIP)'', in the **Federal Register** on September 25, 2003 (68 FR 55416). The notice included a model safety evaluation (SE), a model no significant hazards consideration (NSHC) determination, and a model application.

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, by confirming the applicability of the model NSHC determination to NMP–1 and incorporating it by reference in its application. The model NSHC determination is presented below:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The NRC staff has reviewed the model NSHC determination and its applicability to NMP–1. Based on this review, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: June 6, 2006.

Description of amendment request: The proposed amendments would revise the design basis as described in the Point Beach Nuclear Plant Final Safety Analysis Report (FSAR) by incorporating an updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR part 50, Appendix G, Section IV.A.1.

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

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

The proposed change incorporates the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR part 50, Appendix G, Section IV.A.1 into the FSAR. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with safety analysis assumptions and resultant consequences. Therefore, it is concluded that this change does not significantly increase the probability of occurrence of an accident previously evaluated.

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

The proposed change incorporates the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR part 50, Appendix G, Section IV.A.1 into the FSAR. The change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Would the proposed amendment result in a significant reduction in a margin of safety?

The proposed change incorporates the updated analysis for satisfying the reactor vessel Charpy upper-shelf energy requirements of 10 CFR part 50, Appendix G, Section IV.A.1 into the FSAR. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed change. Therefore, the proposed amendment does not result in a significant reduction in a margin of safety.

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

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

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: May 30, 2006.

Description of amendment request: The proposed amendment would revise the Fort Calhoun Station, Unit 1 (FCS) Technical Specification (TS) requirements related to steam generator 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).

Omaha Public Power District (OPPD) also proposes to change the FCS TS by deleting the sleeving repair alternative to plugging for steam generator tubes. The FCS replacement steam generators (RSGs) to be installed during the fall of 2006 are manufactured by Mitsubishi Heavy Industries, Ltd. (MHI). The change is being requested because OPPD has determined that the sleeving repair alternative to plugging will not be used for the MHI RSGs.

Basis for proposed no significant hazards consideration determination: OPPD stated that it had reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298), as part of the CLIIP. OPPD has concluded that the proposed determination presented in the notice is applicable to FCS and the determination is incorporated by reference to satisfy the requirements of 10 CFR 50.91(a). 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 elimination from the TS surveillance requirements of leak tight sleeves as a repair method alternative to plugging defective steam generator tubes does not introduce an initiator to any previously evaluated accident. The frequency or periodicity of performance of the remaining surveillance requirements for steam generator tubes (including plugged tubes) is not affected by this change. Elimination of the tube repair method has no effect on the consequences of any previously evaluated accident. The proposed changes will not prevent safety systems from performing their accident mitigation function as assumed in the safety analysis.

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

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

Response: No.

The proposed change only affects the TS surveillance requirements. The proposed change is a result of installation of RSGs. The proposed change will eliminate a steam generator tube repair alternative which cannot be utilized or credited for the RSGs. This change will not alter assumptions made in the safety analysis and licensing bases and will not create new or different systems interactions.

Therefore, this 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 deletes surveillance requirements for a steam generator tube repair alternative which will no longer be necessary or applicable. The remaining TS steam generator tube surveillance requirements, including inspection and plugging requirements, will continue to maintain the applicable margin of safety.

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

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao

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 amendment requests: May 30, 2006.

Description of amendment requests: The proposed amendment would revise the Technical Specifications (TSs) 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 includes changes to the TS definition of Leakage, TS 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,'' TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' TS 5.6.10, ''Steam Generator (SG) Tube Inspection Report,'' and adds TS 3.4.17, ''Steam Generator (SG) Tube Integrity.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 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 following NSHC determination in its application dated May 30, 2005.

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

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

A steam generator tube rupture (SGTR) event is one of the design-basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of an 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 a main steamline break (MSLB), rod ejection, and reactor coolant pump locked rotor, the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs are 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 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 an 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 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.

The NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Branch Chief: David Terao.

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

Date of amendment request: May 1, 2006.

Description of amendment request: The proposed amendment would eliminate the requirement for a power range, neutron flux, high negative rate trip and delete the references to this trip as functional Unit 4 in Salem Generating Station (Salem) Unit Nos. 1 and 2 Technical Specification (TS) Table 2.2–1, ''Reactor Trip System Instrumentation Trip Setpoints,'' TS Table 3.3–1, ''Reactor Trip System Instrumentation,'' TS Table 3.3–2, ''Reactor Trip System Instrumentation Response Times,'' and TS Table 4.3–1, ''Reactor Trip System Instrumentation Surveillance Requirements [SRs].'' The proposed changes are consistent with the methodology presented in the Westinghouse Topical Report WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event,'' which has been reviewed by the NRC and found acceptable for referencing in license applications. The amendment also would involve the correction of errata in the TS for Salem Unit Nos. 1 and 2.

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 elimination of the Power Range, Neutron Flux, Negative Rate trip does not increase the probability or consequences of reactor core damage accidents resulting from Rod Cluster Control Assembly (RCCA) Misalignment events previously analyzed. The safety functions of other safety-related systems and components have not been altered. All other Reactor Trip System protection functions are not impacted by the elimination of the requirement for a Power Range, Neutron Flux, High Negative Rate trip. The Power Range, Neutron Flux, High Negative Rate trip circuitry detects and responds to negative reactivity insertion due to RCCA misoperation events, should they occur. Therefore, the Power Range, Neutron Flux, High Negative Rate trip is not assumed in the initiation of such events. The consequences of accidents previously evaluated in the Salem Generating Station (Salem) Updated Final Safety Analysis Report (UFSAR) are unaffected by the proposed changes because no change to any equipment response or accident mitigation scenario has resulted. The proposed changes do not modify the RCCAs or change the acceptance criteria for departure from nucleate boiling (DNB). The TS change reflects analysis described in the UFSAR and cycle-specific analysis performed each fuel cycle.

The proposed revisions to Salem Unit 1 Index page XII, Salem Unit 1 TS 4.2.2.2, Salem Unit 2 TS 4.2.2.2, Salem Unit 1 TS Table 3.3–2, Salem Unit 2 SR number for boron concentration on page 3/4 9-1, Salem Unit 1 TS 6.9.1.5.a, and Salem Unit 1 TS 6.9.1.5.b contain changes administrative in nature that correct errors and do not affect the intent of any TS requirements.

Therefore, the proposed changes do not involve a significant increase in the probability or radiological 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 elimination of the Power Range, Neutron Flux, High Negative Rate trip does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not challenge the performance or integrity of the RCCAs or any other safety-related system. The proposed changes will have no adverse effect on the availability, operability, or performance of the safety-related systems and components assumed to actuate in the event of a design basis accident (DBA) or transient. It has been demonstrated that the Power Range, Neutron Flux, High Negative Rate trip can be eliminated by the NRC approved methodology described in WCAP–11394–P. The Salem fuel cycle specific analyses have confirmed that for a dropped RCCA event, no direct reactor trip or automatic power reduction is required to meet the DNB limits for this Condition II, ''Fault of Moderate Frequency,'' event. The Power Range, Neutron Flux, High Negative Rate trip is not credited either as a primary or backup mitigation feature for any other UFSAR event.

The proposed revisions to Salem Unit 1 Index page XII, Salem Unit 1 TS 4.2.2.2, Salem Unit 2 TS 4.2.2.2, Salem Unit 1 TS Table 3.3–2, Salem Unit 2 SR number for boron concentration on page 3/4 9-1, Salem Unit 1 TS 6.9.1.5.a, and Salem Unit 1 TS 6.9.1.5.b contain changes administrative in nature that correct errors and do not affect the intent of any TS requirements.

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

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

Response: No.

The margin of safety is the difference between the DNB acceptance limit and the failure of the fuel rod cladding. The Salem fuel cycle specific analyses have confirmed that for a dropped RCCA event, DNB limits are not exceeded with the proposed changes. Conformance to the licensing basis acceptance criteria for DBAs and transients with the elimination of the Power Range, Neutron Flux, High Negative Rate trip is demonstrated and the DNB limits are not exceeded when the NRC approved methodology of WCAP–11394–P is applied. The margin of safety associated with the licensing basis acceptance criteria for any postulated accident is unchanged.

The proposed revisions to Salem Unit 1 Index page XII, Salem Unit 1 TS 4.2.2.2, Salem Unit 2 TS 4.2.2.2, Salem Unit 1 TS Table 3.3–2, Salem Unit 2 SR number for boron concentration on page 3/4 9-1, Salem Unit 1 TS 6.9.1.5.a, and Salem Unit 1 TS 6.9.1.5.b contain changes administrative in nature that correct errors and do not affect the intent of any TS requirements.

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

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: May 1, 2006.

Description of amendment request: The amendment would move the main steamline discharge (safety valves and atmospheric dumps) radiation monitors (R46) from the radiation monitoring instrumentation Technical Specification (TS) 3.3.3.1, to the accident monitoring TS 3.3.3.7. The purpose of the R46 monitors is to provide continuous monitoring of high-level, post-accident releases of radioactive noble gases; therefore, relocation to TS 3.3.3.7 is appropriate. In addition, TS definition 1.31, ''Source Checks,'' would be modified to allow different methods to comply with the source check requirement. This change would affect the remaining instruments in TS 3.3.3.1, and would allow for appropriate testing consistent with the technology of the existing detectors, and replacement detectors in the future.

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 the R46 monitors presents no change in the probability or the consequence of an accident, since the monitors are used post-accident for the monitoring of high-level releases of radioactive noble gases.

Relocation of the R46 monitors to the accident monitoring TS 3.3.3.7 is appropriate for the function of the monitors. The R46 monitors are designed to meet the requirements of NUREG–0737 Il.F.1 and the intent of RG [Regulatory Guide] 1.97. The monitor's alarm function is used in the EOPs [Emergency Operating Procedures] to identify a Steam Generator Tube Rupture (SGTR) event EOP entry point and to identify which SG [steam generator] has ruptured. The relocation of the monitor to TS 3.3.3.7 has no affect on the function of the monitor.

The proposed change to the definition of TS 1.31 also does not impact the accident analyses in any manner. The qualitative assessment of monitor response will continue to be performed verifying monitor operability.

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 relocation of the R46 monitors is primarily administrative in nature; there will be no change in the function of the monitors. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Post accident monitoring instrumentation is not associated with the initiation of an accident.

The proposed change to the definition of TS 1.31 also does not create a new or different kind of accident. The qualitative assessment of monitor response will continue to be performed verifying monitor operability.

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

Response: No.

The proposed change to relocate the R46 monitors does not alter the manner in which safety limits, limiting safety systems settings or limiting conditions for operation are determined. The proposed change will not alter any assumptions, initial conditions or results specified in any accident analysis. There is no change in the R46 monitor alarm setpoint.

The proposed change to the TS definition of SOURCE CHECK does not alter the basic requirement that a qualitative assessment of the monitor response be performed; therefore the operability of the monitor will continue to be verified.

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: April 6, 2006.

Description of the amendment request: The proposed amendment changes the existing steam generator (SG) tube surveillance program to one that is consistent with the program proposed by the Technical Specification Task Force (TSTF) in TSTF–449. These changes revise Technical Specification (TS) 1.15, ''Identified Leakage,'' TS 1.21, ''Pressure Boundary Leakage,'' TS 3/4.4.6, ''Steam Generator (SG) Tube Integrity,'' and TS 3/4.4.7.2, ''Operational Leakage,'' and add new administrative TS 6.8.4.i, ''Steam Generator (SG) Program,'' and TS 6.9.1.10, ''Steam Generator Tube Inspection Report.'' Other editorial changes were also made.

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 requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cool down 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.

The structural integrity performance criterion is:

All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

The accident induced leakage performance criterion is:

The primary-to-secondary accident induced leakage rate for any design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm [gallon per minute] per SG.

The operational leakage performance criterion is:

The reactor coolant system operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.

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

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor, the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These analyses 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 retained 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 as part of these TS changes 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 Steam Generator Program required by the proposed addition of TS 6.8.4.i. The program defined by NEI [Nuclear Energy Institute] 97–06 includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

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 Salem TS 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 through 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 that allows SR [Surveillance Requirement] 4.4.7.2.1.d to not be performed until 12 hours after establishment of steady state operation is consistent with NUREG 1431, ''Standard Technical Specifications, Westinghouse Plants'', and ensures the surveillance requirement is appropriate for the LCO [Limiting Condition for Operation].

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 TS 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 TS.

Therefore, the proposed changes do not affect the consequences of an SGTR accident and the probability of such an accident is reduced.

In addition, the proposed changes do not affect the probabilities or consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event.

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 performance based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator 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 Steam Generator 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 changes do 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.

The proposed change that allows SR 4.4.7.2.1.d to not be performed until 12 hours after establishment of steady state operation is consistent with NUREG 1431, ''Standard Technical Specifications, Westinghouse Plants'', and ensures the surveillance requirement is appropriate for the LCO.

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

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

Response: No.

The 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 also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a 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 Steam Generator Program to manage SG tube inspection, assessment, repair and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

The proposed change that allows SR 4.4.7.2.1.d to not be performed until 12 hours after establishment of steady state operation is consistent with NUREG 1431, ''Standard Technical Specifications, Westinghouse Plants'', and ensures the surveillance requirement is appropriate for the LCO.

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

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment requests: June 2, 2006.

Description of amendment requests: The amendment proposes to revise Technical Specification (TS) 3.8.1, ''AC [alternating current] Sources—Operating,'' and TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' to increase the required amount of stored diesel fuel oil to support a change to Ultra Low Sulfur Diesel fuel from California diesel fuel presently in use. This change in the type of fuel oil is mandated by California air pollution control regulations.

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.

This proposed change increases the minimum amount of stored diesel fuel. The change supports the use of Ultra Low Sulfur Diesel (ULSD) fuel rather than the existing California Air Resources Board diesel fuel as mandated by California air pollution control regulations (Title 13 California Code of

Regulations Division 3, Chapter 5, Article 2, Sections 2280–2285).

Technical Specification (TS) 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' requires that each diesel generator have sufficient fuel to operate for a period of 7 days, while the diesel generator (DG) is supplying maximum post Loss of Coolant Accident (LOCA) load demand.

Because the Lower Heating Value (LHV) per gallon of ULSD fuel is less than that of existing diesel fuel, it was necessary to re-calculate the amount of fuel required to supply necessary loads for the required time periods. For Modes 1 through 4, the resulting minimum volumes of ULSD fuel are 48,400 gallons and 41,800 gallons for the 7-day and 6-day fuel supply, respectively. For Modes 5 and 6, the required volumes of ULSD fuel are 43,600 gallons and 37,400 gallons for a 7-day supply and a 6-day supply, respectively.

The DGs and the associated support systems such as the fuel oil storage and transfer systems are designed to mitigate accidents and are not accident initiators. Increasing the minimum volumes of stored fuel in the storage and day tanks will not result in a significant increase in the probability of any accident previously evaluated.

Following implementation of this proposed change, there will be no change in the ability of the diesel generators to supply maximum post-LOCA load demand for 7 days. The proposed minimum volumes of fuel, 48,400 gallons and 41,800 gallons, ensure that a 7-day and [a] 6-day supply of fuel, respectively, are available in Modes 1 through 4. The proposed minimum volumes of fuel, 43,600 gallons and 37,400 gallons, ensure that a 7-day and a 6-day supply, respectively, of fuel is available in Modes 5 and 6. This is identical to the current requirements, except for the increased volume of fuel required due to the decreased heat content of the ULSD fuel. Therefore, this change will not result in a significant increase in the consequences of any accident previously evaluated.

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

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

Response: No.

Following this change, the diesel generators will still be able to supply maximum post-LOCA load demand. The current 7-day and 6-day fuel supply requirements will be maintained following this change. The new required fuel oil volumes are within the capacities of the fuel oil storage tanks.

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

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

Response: No.

The Bases to TS 3.8.3 state that ''[e]ach diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 7 days, while the DG is supplying maximum post loss of coolant accident load demand.'' When the fuel oil tank level is less than required to support the 7-day of operation, the required action depends on whether or not a 6-day supply of fuel is available.

The proposed tank level limits will maintain these 7-day and 6-day fuel supply requirements in all operating Modes following changeout to ULSD fuel.

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 requests involve no significant hazards consideration.

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

NRC Branch Chief: David Terao.

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

Date of amendment request: May 25, 2006.

Description of amendment request: The amendment would revise the Technical Specifications (TSs) to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–372, ''Addition of LCO [Limiting Condition for Operation] 3.0.8, Inoperability of Snubbers.'' The amendment would add (1) a new LCO 3.0.8 addressing when one or more required snubbers are unable to perform their associated support function(s) (*i.e.*, the snubber is inoperable) and (2) a reference to LCO 3.0.8 in LCO 3.0.1 on when LCOs shall be met.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible license amendments adopting TSTF–372 using the NRC's consolidated line item improvement process (CLIIP) for amending licensee's TSs, which included a model safety evaluation (SE) and model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252), which included the resolution of public comments on the model SE. The May 4, 2005, notice of availability referenced the November 24, 2004, notice. The licensee has affirmed the applicability of the following NSHC determination in its application.

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

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

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Allowing delay times for entering [a] supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in [NRC] RG [Regulatory Guide] 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk [, which is required by the proposed TS 3.0.8]. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

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

Date of amendment request: May 25, 2006.

Description of amendment request: The amendment would revise Technical Specifications 3.1.7, ''Rod Position Indication,'' 3.2.1, ''Heat Flux Hot Channel Factor ($F^C_Q(Z)$) (FQ Methodology),'' 3.2.4, ''Quadrant Power Tilt Ratio (QPTR),'' and 3.3.1, ''Reactor Trip System (RTS) Instrumentation.'' The proposed changes are to allow use of the Westinghouse proprietary computer code, the Best Estimate Analyzer for Core Operations—Nuclear (BEACON). The new BEACON power distribution monitoring system (PDMS) would augment the functional capability of the neutron flux mapping system for the purposes of power distribution surveillances at the Callaway Plant. Certain required actions, for when a limiting condition for operation is not met, and certain surveillance requirements are being changed to refer to power distribution measurements or measurement information of the core.

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 PDMS performs continuous core power distribution monitoring with data input from existing plant instrumentation. This system utilizes an NRC-approved Westinghouse proprietary computer code, *i.e.*, Best Estimate Analyzer for Core Operations µ Nuclear (BEACON), to provide data reduction for incore flux maps, core parameter analysis, load follow operation simulation, and core predication. The PDMS does not provide any protection or control system function. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Final Safety Analysis Report (FSAR) [for the Callaway Plant].

Use of the PDMS supports maintaining the core power distribution within required limits. Further continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (*i.e.*, earlier determination of an adverse condition developing) for operation action prior to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

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

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

Response: No.

Other than use of the PDMS to monitor core power distribution, implementation of the PDMS and associated Technical Specification changes has no impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation [other than core power distribution monitoring] will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with [the] implementation of the PDMS do not result in a change to the design basis of any plant component or system [other than to the PDMS]. The evaluation of the effects of using the PDMS to monitor core power distribution parameters shows that all design standards and applicable safety criteria limits are met. [The PDMS is to monitor the core power distribution and is, therefore, not an accident initiator.]

The proposed changes do not result in any event previously deemed incredible being made credible [by the implementation of the PDMS]. Implementation of the PDMS will not result in any additional adverse condition and will not result in any increase in the challenges to safety systems. The cycle-specific variables required by the PDMS are calculated using NRC-approved methods. The Technical Specifications will continue to require operation within the required core operating limits, and appropriate actions will continue to be [required to be] taken when or if limits are exceeded.

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

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

Response: No.

No margin of safety is adversely affected by the implementation of the PDMS. The margins of safety provided by [the] current Technical Specification requirements and limits remain unchanged, as the Technical Specifications will continue to require operation within the core limits that are based on NRC-approved reload design methodologies. [These NRC-approved reload design methodologies are not being changed.] Appropriate measures exist to control the values of these cycle-specific limits, and appropriate actions will continue to be specified and [required to be] taken for when limits are violated. Such actions remain unchanged.

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

The NRC staff has reviewed the licensee's analysis and, based on 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., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

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

Date of amendment request: June 2, 2006.

Description of amendment request: The amendment would revise Surveillance Requirement 3.5.2.8 in the Technical Specifications by replacing the phrase ''trash racks and screens'' with the word ''strainers.'' The amendment reflects the replacement of the containment sump suction inlet trash racks and screens with a complex strainer design with significantly larger effective area in the upcoming Refueling Outage 15. This is in response to Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'' dated September 13, 2004.

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 consequences of accidents evaluated in the Updated Safety Analysis Report (USAR) [for the Wolf Creek Generating Station] that could be affected by the proposed change are those involving the pressurization of containment and associated flooding of the containment and recirculation of this fluid within the Emergency Core Cooling System (ECCS) or the Containment Spray System (CSS) (*e.g.*, Loss of Coolant Accidents). The proposed change does not impact the initiation or probability of occurrence of any accident. [The containment sump trash racks and screens, and the sump strainers that are replacing the trash racks and screens are not initiators of accidents.]

Although the configurations of the existing containment recirculation sump trash racks and screen[s,] and the replacement sump strainer assemblies are different, they serve the same fundamental purpose of passively removing debris from the sump's suction supply of the supported system pumps. Removal of trash racks does not impact the adequacy of the pump NPSH [net positive suction head] assumed in the safety analysis. Likewise, the change does not reduce the reliability of any supported systems or introduce any new system interactions. The greatly increased surface area of the new strainer is designed to reduce head loss [at the containment sump] and reduce the approach velocity at the strainer face significantly, decreasing the risk of impact from large debris entrained in the sump flow stream.

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 containment recirculation sump strainers are a passive system used for accident mitigation. As such, they cannot be accident initiators. Therefore, there is no possibility that this change could create any new or different kind of accident.

No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of the proposed change. There will be no adverse effect[s] or challenges imposed on any safety related system as a result of the change. Therefore, the possibility of a new or different type of accident is not created. [The containment recirculation sump suction inlet trash racks and screens are being replaced with a complex strainer design with significantly larger effective surface area to reduce head loss and reduce the approach velocity at the strainer face significantly, decreasing the risk of impact from large debris entrained in the sump flow stream.]

There are no changes which would cause the malfunction of safety related equipment, assumed to be OPERABLE in the accident analyses, as a result of the proposed Technical Specification change. No new equipment performance burdens are imposed. The possibility of a malfunction of safety related equipment with a different result [or consequences] is not created.

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 does not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. The radiological dose consequence acceptance criteria listed in the Standard Review Plan [for accidents] will continue to be met.

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: Jay Silberg, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

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.*

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

Date of application for amendments: March 9, 2005, as supplemented by letter dated July 7, 2005.

Brief description of amendments: The amendments revised the Millstone Power Station, Unit Nos. 2 and 3 Technical Specifications to incorporate wording related to the reactor coolant system, electrical power system and refueling operations to provide operational flexibility during mode changes or addition of coolant during shutdown operations.

Date of issuance: June 28, 2006.

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

Amendment Nos.: 293 and 230.

Facility Operating License Nos. DPR–65 and NPF–49: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29788). The additional information provided in the supplemental letter dated July 7, 2005, did not expand the scope of the application as noticed and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 28, 2006.

No significant hazards consideration comments received: No.

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

Date of application of amendments: January 5, 2005, as supplemented November 21, 2005.

Brief description of amendments: The amendments revised Technical Specifications (TSs) 5.5.19.b, 5.1.19.c, and TS Surveillance Requirement (SR) 3.8.1.9 associated with the Lee Combustion Turbine (LCT) testing program. TS 5.5.19 required verification that an LCT can supply the equivalent of one unit's maximum safeguards loads, plus two units' Mode 3 loads when connected to the system grid every 12 months. The amendments clarified this requirement as ''Verify an LCT can supply equivalent of one unit's Loss of Coolant Accident (LOCA) loads plus two units' Loss of Offsite Power (LOOP) loads when connected to system grid every 12 months.'' TS 5.5.19.c and SR 3.8.1.9 were revised for consistency.

Date of Issuance: July 5, 2006.

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

Amendment Nos.: 352/354/353.

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

Date of initial notice in ***Federal Register:*** February 15, 2005 (70 FR 7764). The additional information provided in the supplemental letter dated November 21, 2005, did not expand the scope of the application as noticed and did not change the NRC staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 29, 2005.

Brief description of amendment: The amendment deleted License Condition, Section 2.F, that requires the reporting of violations in Section 2.C of the Facility Operating License.

Date of issuance: June 28, 2006.

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

Amendment No.: 116.

Facility Operating License No. NPF–69: Amendment revised the Facility Operating License.

Date of initial notice in ***Federal Register:*** April 25, 2006 (71 FR 23958).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 28, 2006.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Appling County, Georgia

Date of application for amendments: February 17, 2006.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) adding Limiting Condition for Operation (LCO) 3.0.8 to allow a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4).

Date of issuance: June 29, 2006.

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

Amendment Nos.: 250/194.

Renewed Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register:*** April 25, 2006 (71 FR 23960).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 17, 2006.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) adding Limiting Condition for Operation (LCO) 3.0.8 and renumbering existing LCO 3.0.8 to LCO 3.0.9 to allow a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4).

Date of issuance: June 29, 2006.

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

Amendment Nos.: 141/121.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Licenses and the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 25, 2006 (71 FR 23960).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 19, 2005, as supplemented by letter dated March 30, 2006.

Brief description of amendments: The amendments modified several parts of Technical Specification Surveillance Requirement (SR) 4.0.5, both to change the surveillance intervals for which the 25 percent extension provided in SR 3.0.2 would apply, and to replace the references in SR 4.0.5 to the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code,* Section XI, with the ASME Operation and Maintenance Code.

Date of issuance: June 16, 2006.

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

Amendment Nos.: 308 and 297.

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

Date of initial notice in ***Federal Register:*** February 14, 2006 (71 FR 7183).

The supplemental letter dated March 30, 2006, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 16, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 20, 2006, as supplemented on May 15, 2006.

Brief description of amendments: These amendments revised the reactor coolant pressure and temperature limits, low-temperature overpressure protection system (LTOPS) setpoint values, and LTOPS enable temperatures for up to 28.8 effective full-power years (EFPYs) and 29.4 EFPYs of operation at Surry Power Station, Unit Nos. 1 and 2, respectively.

Date of issuance: June 29, 2006.

Effective date: As of the date of issuance.

Amendment Nos.: 248/247.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments revised the License and the Technical Specifications.

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

The May 15, 2006, supplement contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 11th day of July.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. 06–6246 Filed 7–17–06; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting

FEDERAL REGISTER CITATION OF PREVIOUS ANNOUNCEMENT: [71 FR 40174, July 14, 2006].

STATUS: Closed meeting.

PLACE: 100 F Street, NE., Washington, DC.

DATE AND TIME OF PREVIOUSLY ANNOUNCED MEETING: Tuesday, July 18, 2006 at 10 a.m.

CHANGE IN THE MEETING: Time change.

The closed meeting scheduled for Tuesday, July 18, 2006 at 10 a.m. has been changed to Tuesday, July 18, 2006 at 11 a.m.

At times, changes in Commission priorities require alterations in the scheduling of meeting items. For further information and to ascertain what, if any, matters have been added, deleted or postponed, please contact the Office of the Secretary at (202) 551–5400.

Dated: July 14, 2006.

J. Lynn Taylor,

Assistant Secretary.

[FR Doc. 06–6303 Filed 7–14–06; 10:52 am]

BILLING CODE 8010–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–54136; File No. 4–517]

Program for Allocation of Regulatory Responsibilities Pursuant to Rule 17d–2; Order Granting Approval of Plan for Allocation of Regulatory Responsibilities Between The NASDAQ Stock Market LLC and the National Association of Securities Dealers, Inc.

July 12, 2006.

Notice is hereby given that the Securities and Exchange Commission (''SEC'' or ''Commission'') has issued an Order, pursuant to Sections 17(d) [1] and 11A(a)(3)(B) [2] of the Securities Exchange of 1934 (''Act''), granting approval and declaring effective a plan for allocating regulatory responsibility filed pursuant to Rule 17d–2 of the Act,[3] by The NASDAQ Stock Market LLC (''Nasdaq'') and the National Association of Securities Dealers, Inc. (''NASD'').

Accordingly, NASD shall assume, in addition to the regulatory responsibility it has under the Act, the regulatory responsibilities allocated to it under the plan. At the same time, Nasdaq is relieved of those regulatory responsibilities allocated to NASD.[4]

I. Introduction

Section 19(g)(1) of the Act,[5] among other things, requires every national securities exchange and registered securities association (''SRO'') to examine for, and enforce compliance by, its members and persons associated with its members with the Act, the rules and regulations thereunder, and the SRO's own rules, unless the SRO is relieved of this responsibility pursuant to Section 17(d) or 19(g)(2) of the Act.[6] Section 17(d)(1) of the Act was intended, in part, to eliminate unnecessary multiple examinations and regulatory duplication for those broker-dealers that maintain memberships in more than one SRO.[7] With respect to common members of two or more SROs, Section 17(d)(1) authorizes the Commission, by rule or order, to relieve an SRO of the responsibility to receive regulatory reports, to examine for and enforce compliance with applicable statutes, rules and regulations, or to perform other specified regulatory functions.

To implement Section 17(d)(1), the Commission adopted two rules: Rule 17d–1[8] and Rule 17d–2 under the Act.[9] Rule 17d–2 under the Act permits SROs to propose joint plans allocating regulatory responsibilities, other than financial responsibility rules, with respect to common members. Under paragraph (c) of Rule 17d–2, the Commission may declare such a plan effective if, after providing for notice and comment, it determines that the plan is necessary or appropriate in the public interest and for the protection of investors, to foster cooperation and coordination among self-regulatory organizations, to remove impediments to and foster the development of a national market system and a national clearance and settlement system, and in conformity with the factors set forth in Section 17(d) of the Act. Upon effectiveness of a plan filed pursuant to Rule 17d–2, any self-regulatory organization is relieved of those regulatory responsibilities for common members that are allocated by the plan to another self-regulatory organization.

On April 17, 2006, the Commission published notice of the filing by Nasdaq and NASD of a joint plan allocating regulatory responsibility for common members.[10] No comments were received. On July 12, 2006, Nasdaq and NASD filed an amended joint plan for

[1] 15 U.S.C. 78q(d).

[2] 15 U.S.C. 78k–1(a)(3)(B).

[3] 17 CFR 240.17d–2.

[4] On January 13, 2006, the Commission approved Nasdaq's application for registration as a national securities exchange. The Commission conditioned the operation of the Nasdaq Exchange upon satisfaction of several requirements, one of which was the approval by the Commission of an agreement pursuant to Rule 17d–2 between Nasdaq and NASD. Securities Exchange Act Release No. 53128, 71 FR 3550 (January 23, 2006). Commission approval of this plan allocating regulatory responsibility satisfies this requirement.

[5] 15 U.S.C. 78s(g)(1).

[6] 15 U.S.C. 78q(d) and 15 U.S.C. 78s(g)(2).

[7] Securities Acts Amendments of 1975, Report of the Senate Committee on Banking, Housing, and Urban Affairs to Accompany S. 249, S. Rep. No. 94–75, 94th Cong., 1st Session. 32 (1975).

[8] 17 CFR 240.17d–1. Rule 17d–1 authorizes the Commission to designate a single SRO as the designated examining authority (''DEA'') to examine common members for compliance with financial responsibility requirements imposed by the Act, the rules thereunder, and SRO rules.

[9] 17 CFR 240.17d–2.

[10] Securities Exchange Act Release No. 53628 (April 10, 2006), 71 FR 19763.